NUCLEAR SAFETY

A REVIEW OF RECENT DEVELOPMENTS

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foreword

This quarterly journal is one of a series of Technical Progress Reviews prepared by the Oak Ridge National Laboratory at the request of the Division of Information Services, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter, but rather a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered the province of health physics. Even with these exclusions, nuclear safety cuts across such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, where given, represent the opinions of the editors, who are employed by the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information. Many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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NUCLEAR SAFETY

NUCLEAR SAFETY CRITERIA

AEC Reactor Safety Program

In this country the design, development, and operation of nuclear reactors covers a period of over 15 years. During this time no known injury to the public has occurred through the operation of a reactor.

This remarkable safety record in an infant industry, with its share of potential hazards, has resulted from many factors. Of paramount importance is the safety consciousness of the reactor designer. However, of inestimable importance are AEC regulations governing licensing, operation, and periodic inspections of reactors and AEC's technical program of reactor safety research.

Designers are, of course, thoroughly familiar with the degree of safety which must be demonstrated to obtain AEC approval for building and operating a reactor or any major research facility in which a critical mass can be attained. Not everyone in the nuclear energy field is so familiar, however, with AEC's technical program of reactor safety research. Therefore, the AEC has issued a book[1] summarizing the safety research and development program and illustrating the diversity of the problems involved. The approach to reactor safety exemplified by this research program has a broader base than if it were only a part of each new reactor program.

The primary basis for the program is the study of two postulated accidents that could lead to core meltdown and subsequent fission-product release; (1) loss of nuclear control and (2) loss of coolant during power operation.

Since either of these accidents could result in severe damage and, possibly, breaching of the reactor pressure vessel, studies of the effects of these accidents are the primary activities. The program is categorized by the three activities discussed below. Each discussion is preceded by a brief statement of the program in the form of a quotation from the Commission's book.[1]

Kinetics and Control

> The dynamic characteristics of reactors are studied to gain an understanding of abnormal reactor behavior. This includes the experimental investigation of reactors under transient and simulated runaway conditions to obtain information which may be used to develop analytical techniques for ensuring safe reactor design. The development of reactor control systems and of special safety devices also forms a part of this work.

The kinetics and control program involves theoretical studies; reactor experiments performed with the SPERT (Special Power Excursion Reactor Test) facility, the KEWB (Kinetic Experiments on Water Boilers) facility, and the EBR (Experimental Breeder Reactor); safety-system performance tests; and safety-device development and testing. The currently available reports on these activities are listed in reference 1. The published literature on safety devices was reviewed in the previous issue of this Review.[2]

Potential Reactions

> Studies of certain secondary reactions which could be started by a reactor incident are included in this category. Among the most important of these is an understanding of metal-water reactions, metal-gas reactions, and fission-product release from core meltdown.

In the potential reactions program, investigations of fission-product release, atmospheric dispersion of fission products, metal-water reactions, and metal-gas reactions are under way.

The Transient Reactor Test Facility (TREAT) is being used to determine the effect of meltdown on reactivity. Reference 1 lists 13 reports on these projects. The published literature on fission-product release was reviewed in the previous issue of this Review,[3] and some of the material on chemical reactions is discussed in Sec. V of this issue.

Containment

In this phase of the experimental safety program, structures designed to prevent the release of fission products to the environment are studied. The dynamic behavior of reactor pressure vessels and vapor containment structures is being studied under various stress conditions imposed by simulated reactor incidents.

The containment program includes studies of the forces that would be generated in an accident and their effects. Information being obtained includes energy sources, pressure transmission, missile damage, and effect of earthquakes. The containment vessels that would be needed in the event of an accident are being considered. The effectiveness of blast shields is being studied, as well as the deformation and rupture of pressure vessels and the elastic response of such shells. Twelve reports on the projects in this category are listed in reference 1. Some recent literature on the integrity of containment vessels was reviewed in the previous issue of this Review,[4] and the pertinent available literature on the effects of earthquakes on reactor design is discussed in Sec. III of this issue.

(T. E. Cole)

Site Selection Criteria

The problem of selecting a suitable site for the location of a nuclear reactor has long plagued reactor engineers. As discussed in the previous issue of this Review,[5] the AEC is preparing criteria for the evaluation of proposed sites for nuclear reactors. Concurrently, subcommittees of both the American Nuclear Society (ANS) Standards Committee and the American Standards Association (ASA) have been concerned with the same problem. A progress report of the efforts of the ANS Standards Subcommittee on the formulation of site criteria was presented by R. O. Brittan of Argonne National Laboratory at the ANS meeting held in Gatlinburg, Tenn., June 15, 1959. The ANS Standards Subcommittee coordinates its work with the work of the ASA as Subcommittee No. 1 of the ASA Sectional Committee No. 6.

The first task undertaken by the Subcommittee was the preparation of a procedural standard to guide consideration of the influence of abnormal reactor behavior on site evaluation and to recommend means of specifying allowable leakage rates and auxiliary shielding requirements for contained stationary systems producing sensible power. ("Sensible" power here refers to systems in which most of the fissions occur during normal operation rather than in an excursion burst of fissions.) A preliminary draft of this procedural standard has been submitted to the sectional committee for consideration. The following is a summary of the scope and procedures followed in this first task, as described by Brittan.

The procedure is proposed for use in nuclear reactor plants provided with a containment barrier that is not impaired by the reactor accident. Such a container will have associated with it a leakage rate that can be used to define the source strength of the leaking activity. To the extent that direct gamma radiation from the containment vessel following the accident may contribute to the exposure of off-site personnel, the biological shielding is also an important factor.

The procedure next considers the problem of the evaluation of the exposure to the public as a consequence of the accident. Such an evaluation includes the determination of (1) the radioactive material inventory of the reactor, (2) the fractional release and disposition of this inventory within the containment vessel, (3) the time-dependent overpressure within the containment vessel, and (4) the fraction of the inventory which can escape from the containment vessel.

Up to this point the analysis is independent of the site, assuming that the leakage rate is fixed. It is then necessary to evaluate the consequences of the release of activity on the surrounding populace and land to determine the control boundary required to ensure against injury. Conversely, a permissible leakage rate can be determined for a preselected control-boundary location.

The preceding factors dictate that a controlled site be selected in such a manner that its size and arrangement preclude the credible possibility of excessive damage to the public in the

form of radiation injury or contamination. The balance finally obtained between these factors and site size and location is always resolved on an economic basis. The guide does not purport to strike such a balance, but it provides procedures for considering each of these components.

After a report on the progress of the committee, Brittan continued with a discussion of some of the criticisms, objections, disagreements, and unresolved difficulties encountered. A condensed version of some of the more important points follows:

> The basic question of what to use as a damage criterion has been raised, as it should have been. The guide, in its first draft form, deliberately chose to set the criterion of *no* injury beyond the controlled boundary, so that debate over this point might lead to correct or reasonable selection of criteria compatible with the public philosophy regarding risks. I personally feel that, in view of the infrequency of severe accidents likely to occur, it would be best to select criteria which would place the risk in the same ball park as for nonatomic industrial plants, for which operational and accident experience is available over many years. Following up this theme, it appears that the American public continues to allow industrial complexes to be built and to operate even when accident experience shows that disasters resulting in the death of over 500 people and injury in varying degree to several thousand can and have occurred. One could use as a damage criterion a radiation dose [vs.] population distribution number that would be less than the product of a killing dose and 500 people. This product would be 200,000 if one used a killing dose of 400 r, a reasonable LD_{50} number . . . .
>
> There is some objection to using only the most unfavorable meterology in proceeding with application of the criterion to a specific case . . . .[*] I would like to suggest rather that the criteria be so stated as to allow the person wrestling with this problem to work out numbers on the basis of acceptable proof that the meteorology of the site being considered is always more favorable than the most extreme conditions. This would not allow departure from the general procedure outlined but would credit and recognize the desirability of realism in handling the problem. Such an allowance should be made not only in [the] case of meteorology, but generally throughout the procedure . . . .
>
> A further criticism encountered is that only the continuous release case is handled . . . .[*] The case of the uncontained system was not intended to be considered because, for the most pressing needs of the power reactor industry, such a case would not be reasonable . . . . Furthermore, the continuous release case can be used to approximate the puff by mathematical treatment . . . .
>
> In lieu of considering each isotope separately, certain groupings were recommended . . . . The proposed grouping would be (1) noble gases, (2) halogens, (3) volatile solids, and (4) nonvolatile solids.

The ANS Standards Committee has properly identified the problems inherent in the evaluation of a reactor site and has observed that exclusion area and leakage rate are dependent parameters. At its present state of formulation, however, the Committee's report has not yet designated the exposure criteria that should be attained at the site boundaries.

A recent Oak Ridge National Laboratory report[8] notes that, in addition to limiting the maximum permissible individual total-body exposure, the genetic dose (average dose per person from conception to 30 years of age) to the population must be limited and suggests values of 25 and 0.3 r, respectively, for these two exposures. The National Committee on Radiation Protection[9] (NCRP) and the International Commission on Radiological Protection[10] (ICRP) recommend that an emergency total dose of up to 25 rem, occurring only once in a lifetime, not affect the exposure status of the individual.

The total population exposure is derived in part from recommendations of the ICRP[10] which state that the permissible genetic dose should not exceed 5 rem (exclusive of natural background and medical exposure, which, in the United States, average 4 and 5 r, respectively). No firm recommendations for apportioning the 5 rem are made by the ICRP, but that Commission gives the following apportionment as an illustration:

	Dose, rem
Occupational exposure groups	1.0
Exposure of special groups (principally the public living in the neighborhood of reactors)	0.5
Exposure of the population at large	2.0
Reserve	1.5

In a paper[11] presented at the 1959 Health Physics Society Annual Meeting, C. R. McCullough, Chairman, AEC Advisory Committee for Reactor Safeguards, proposed 2,000,000 man-rem as

*For a more comprehensive treatment of these questions, references 6 and 7 are suggested as good intermediate sources, which, in turn, refer to detailed analyses.

the limit on the population exposure from a single reactor maximum credible accident. If it is assumed that there may eventually be one such accident every year and that the population is 2×10^8, the resulting genetic dose per person in 30 years is

$$1 \times \frac{2 \times 10^6 \times 30}{2 \times 10^8} = 0.3 \text{ r}$$

This dose is appreciably less than that allocated by the ICRP for the population at large or the reserve. However, since significant portions of these allocations may be required for routine public exposures, the selection of a value as low as 0.3 r has considerable merit for planning purposes, although it is somewhat conservative at the present time.

This population exposure may be compared with that recently proposed by the British Medical Research Council[12] for application in Great Britain, in which a value of 10 r total-body exposure would be permitted in instances where the number of exposed persons is small compared with one-fiftieth of the total population. "Small" is not defined in this report, but the calculated genetic dose to age 30 for one-fiftieth of the population to receive 10 r would be 6 r if one accident per year in a population of 5×10^7 were assumed. If 10 per cent may be considered a small percentage, the resulting genetic dose would still be twice that proposed by McCullough, even though the emergency exposure specified by the British is a factor of 2.5 less than the recommended NCRP value.

In a paper presented at the International Symposium on the Safety and Location of Nuclear Plants, Farmer and Fletcher[13] of the United Kingdom Atomic Energy Authority discussed specific siting criteria. This paper is one of the more comprehensive treatments of this subject, although the subject matter is discussed in terms of a specific activity release and, consequently, is not developed as fully as would be desired for general application. Farmer and Fletcher have considered the consequences of both routine and accidental releases of activity. They specify that the reactor plant should be so designed that the radiation exposure from all sources (direct, inhalation, and ingestion) to the people (including children) living in the neighborhood of a controlled area will not exceed 0.5 rem/year, as recommended by the ICRP.[10] The criteria for exposure following a major accident are derived from the recommendations of the British Medical Council[12] which propose maximum individual exposures to the thyroid, bone, and total body of 25, 15, and 10 rads, respectively. For the assumed release of 250 curies of iodine and 0.1 curie of strontium (the ratio of iodine to strontium release was derived from the Windscale accident),[14] it is possible to determine, with the use of Sutton's equations for typical meteorological conditions, the various areas surrounding the site from which persons should be evacuated and from which the consumption of foodstuffs and milk would have to be controlled. The I^{131} exposure is predominant, and, not surprisingly, the exposure of the children in the population to I^{131} was the most restrictive criterion.

In addition to the maximum individual exposure, the desirability of restricting the number of persons thus exposed was recognized by Farmer and Fletcher. Toward this end, but seemingly rather arbitrarily, they propose that: "(1) few persons should live within 500 yd of the site; (2) in any 10° sector around the site, not more than 500 people should live within 1.5 miles; and (3) no large center of population of 10,000 or more should lie within 5 miles."

Although the report of Farmer and Fletcher does not discuss reactor containment as such, it is apparent that any modification which would change the source strength would also change the areas and distances that are given in this report. The use of a 0.5 rem/year limit on routine exposure to off-site personnel is consistent with practice in this country, although the maximum individual total-body accidental exposure recommended by the British Medical Council is 10 rads, compared with a value of 25 rads proposed by the NCRP.[9] The emergency values for thyroid and bone exposures recommended by the British Medical Council are likewise lower than comparable values derived from the NCRP data. These differences will ultimately be resolved by the various national and international health physics organizations. Perhaps the most serious shortcoming of the paper by Farmer and Fletcher[13] is the omission of generalized criteria for establishing the allowable surrounding population, and it is not obvious that the surrounding population exposure should include the exposure from the ingestion of contaminated foodstuffs and milk. The correlation of the population exposure to some allowable genetic dose, as proposed by

ORNL[8] and by McCullough,[11] would seem to be the basic population exposure criterion.

(W. B. Cottrell)

Safety in the Handling of Fissionable Materials

Care must be taken in manufacturing, processing, packaging, and shipping fissionable materials to avoid critical concentrations. The concentration limits are a function of the substance in question, the total amount of the material, and, in particular, geometry. For a number of fissionable materials and compounds of general interest, tables have been prepared listing the allowable safe concentrations. In instances where it is not practical to follow these existing guides, which establish the requirements for geometric control of the fissionable material, reliance must be placed on administrative control. Such control is not otherwise preferred to geometric control, which is less subject to human fallibilities. It is not entirely coincidental that the only criticality accidents in assemblies which were never intended to approach criticality (the Los Alamos incident[15] and the Y-12 incident (Sec. VI)) occurred in processing areas which were under administrative control.

Transportation Problems

The greatest hazard to the public in the handling of fissionable materials arises from the transporting of these materials to and from manufacturing and processing sites and the locations at which they are to be used. Federal regulations[16,17] specifically limit the quantities of potentially chain-reacting materials which may be transported by unlicensed carriers or by licensees in their own vehicles, and thus the probability of a criticality accident in the transportation of these materials is greatly reduced. The limits established by the regulations are

	Maximum amount, g	
	For delivery to a carrier	For transportation by a licensee
U^{235} content of enriched uranium	100	350
U^{235}	60	200
Plutonium	60	200

Shipments in excess of the prescribed limits can be made if the proposed shipping procedures have received AEC approval. Once such approval has been obtained, the licensee may make any number of such shipments in the approved manner without approval of each shipment. Since the report cited in reference 17 was prepared for the attention and information of legal groups, specific identification of the source of the information would have been helpful. It is the opinion of the reviewers that reference 17 is a reasonable interpretation of the regulations cited in reference 16.

Although nearly all shipments of production quantities of uranium materials are made in "exclusive use" vehicles, the civilian and foreign reactor programs and the AEC requirements for the shipment of smaller quantities of uranium of many different U^{235} enrichments created a need for the development of methods for shipping enriched uranium that would be more economical than shipment in exclusive-use vehicles and, at the same time, afford assurance of being safe from a criticality standpoint. To fulfill this need, a working committee, consisting of representatives of the AEC and operating contractors under the direction of the Oak Ridge Operations Office, was appointed in 1957 to prepare a "Guide to Shipment of U^{235} Enriched Uranium Materials"[18] which would define nuclearly safe methods of shipping enriched-uranium materials.

The provisions of the Guide are intended to be applicable only to systems containing uranium materials enriched in U^{235}, and the considerations made involve only the nuclear safety aspects of such materials. Thus, the hazards of fire or explosion, the hazards that might result from insufficient structural strength of containers or bracing, and the toxicity or radioactivity of some of the materials or their compounds are not considered, although their importance in the over-all nuclear safety picture is recognized.

The tables of permissible values are based on (1) criticality experiments actually performed or (2) conservative interpolations of these experiments. The nuclear safety factors incorporated into the criteria and suggestions of the Guide are based primarily on the experience and information developed by those who have been in charge of nuclear safety in their own plant operations, with the aim that methods of shipment will not only be safe from a nuclear

standpoint but will also be as economical as possible, consistent with nuclear safety. To assist in solving problems, the Guide outlines and illustrates possible packaging, handling, and shipping procedures which are considered to provide a reasonable degree of nuclear safety. Detailed specifications of some containers and packaging arrangements that have proved to be effective in the shipment of various specific uranium-containing materials are given. The Guide also specifies the conditions under which shipments may be made outside the limitations, provided that the over-all nuclear safety of the particular shipments will not be decreased.

General criteria relative to the types of uranium-containing materials to be shipped, methods of packaging and spacing, means for preventing interaction, and emergency handling procedures are presented. The basic nuclear safety requirements for various uranium systems, such as dry salts and solutions of various densities or concentrations, metal, and uranium-metal alloys, are summarized. A glossary of some of the special terms used in the Guide is included, as well as scrap material shipment information, formulas for computing interaction solid angles, and other items of interest.

(A. J. Mallett)

Manufacture and Processing

In manufacturing and processing fissionable materials, the emphasis changes from public safety to occupational safety. Manufacture and/or processing may involve purification, separation, reclamation, disposal, or other operations with fissionable materials. Since the processing methods and precautions required differ with different materials and different processes, the processes used at only one plant, the Oak Ridge Gaseous Diffusion Plant (ORGDP), have been selected for review. It was found that condensation of tables and graphs for even a single facility was inadvisable because of the danger of misinterpretation of out-of-context information. However, because of the value of an insight into the scope of the task of specifying the detailed design and operation of a large facility, the document[19] entitled "Criticality Data and Nuclear Safety Guide Applicable to the Oak Ridge Gaseous Diffusion Plant" is reviewed here briefly.

The materials covered in the report include various uranium compounds and their solutions or water mixtures, as well as UF_6 in its various states. This report[19] is the fifth revision of a pioneer report[20] that was issued in 1951, in which a first attempt was made to organize the then available critical mass data and to develop a consistent set of nuclear safety rules for the prevention of accidental nuclear chain reactions.

As in the previous editions, one of the principal aims of the current report has been the tabulation of the various minimum critical conditions, both experimental and theoretical, from which suitable nuclear safety criteria may be derived for the low-density uranium systems, including, in particular, the solutions encountered at ORGDP. New criticality data include the results of experiments with uranium at U^{235} enrichments of 1.02, 2.0, and 44.6 per cent. Since all data have been carefully documented, the extensive bibliography should prove particularly valuable to those persons interested in the field of nuclear safety, as should those sections on the nuclear fundamentals of equipment design, applied design methods and techniques, and basic operational philosophy. A condensed glossary of nuclear safety terms and a brief outline of the administrative organization for nuclear safety control at ORGDP are also included in the report.

The basic nuclear safety criteria remain essentially unchanged from those given in previous issues of the report. As a reflection of experimental and theoretical developments, however, the U^{235} enrichments at which ORGDP materials are considered to be nonreactive have been increased from 0.71 to 0.90 per cent, a new mass-volume principle has been included, and values of the nuclearly safe variables for U^{235} enrichments below 5 per cent have been extended. The document summarizes the available information and its use in numerous tables and graphs. The criteria given include basic assumptions concerning the uranium materials processed at ORGDP, fundamentals of equipment design, operational controls for handling uranium materials in the gaseous, liquid, or solid phases at various U^{235} enrichments and moderations, and use of nuclear poisons. Tables give safe geometrical and mass variables, describe safe pipe connections, and list safe spacing for typical ORGDP containers. Formulas for calculating safe variables at intermediate densities, as well as a sample chart for computing safe solid angles for use with the interaction criteria developed at ORGDP, are also included.

Although the report[19] was prepared specifically for gaseous diffusion plant operations, it has proved to be of general interest in the field of nuclear safety. Therefore it becomes important to reiterate that the criteria presented are considered to be primarily applicable for the uranium materials normally encountered in diffusion plant operations, as enumerated under the section of Basic Assumptions. Perhaps the most important of these limitations are those on uranium density, which is assumed to be ≤ 3.2 g of uranium per milliliter of solution, and on types of moderators and reflectors available, which are assumed to be no better than ordinary water. Thus, although the methods outlined in the report may be quite generally applicable, the conclusions drawn are not necessarily compatible with the local conditions or materials that may be encountered in other production or laboratory facilities. Caution should therefore be excerised in the use of these criteria for high-density U^{235} materials, and, obviously, they should not be used at all for other fissionable materials, such as U^{233} or plutonium. Source information on these other fissionable materials is available in other summary reports.

(C. E. Newlon)

The editors recognize the frustration of attempting to locate a complete "safety manual" for the prevention of criticality accidents or to locate criteria for the development of procedures which are safe and inexpensive. On the bright side, however, the remarkably good record of freedom from criticality accidents in processing and shipping fissionable materials does show that accident experience is not a requirement for accident prevention in this field.

(L. A. Mann)

Law and Atomic Energy Hazards

The effects of laws, treaties, conventions, and other political structures on the development and use of atomic energy are so profound that these legal "institutions" are rapidly assuming an importance as great as the recent technological advances in atomic energy. Therefore it is of extreme importance that local, national, and international legal and political structures be sound, progressive, and intelligent. It is also certain in today's world that the associated problems will continue to be great.

Although no attempt is made to present a complete review of the whole structure of the interrelation of the field of law with the hazards of atomic energy, several recent articles that give a good insight into the general problem are reviewed here. These articles include consideration of local and international activities in this area, as well as several articles on liability and indemnification on national and international levels and the problem of proving radiation damage.

(L. A. Mann)

International Legal Problems

In a paper presented at the 37th Annual Meeting of the American Branch of the International Law Association, Von Mehren[21] points out that the development of nuclear power is an international endeavor which has created and will create problems of an international character and which has already caused a "remarkable development of international institutions." The problems not only concern the scientist and the engineer but also the statesman, the politician (political scientist), and the lawyer; "little will be gained if the scientific and engineering problems are solved while the governmental and legal problems remain unsolved."

Von Mehren discusses "(1) the problems created by the fact that science and technology have outpaced the law, (2) the problems created by the failure to coordinate international growth in the atomic field, and (3) the problems of a traditional nature posed by the atom in the international field." In concluding that science and technology have outpaced law, Von Mehren finds that "the law has no laboratory save the community" and "it must gather its experience in the hurly-burly of everyday life rather than within the walls of the university or industrial laboratory." Although "man had a millennium in which to accustom himself and to shape his institutions to the requirements of a technology based on fire as his principal source of energy," the history of atomic fission as it relates to man's search for a new source of energy is very different. Today, man "faces the problem of adjusting himself and his institutions within a very short period of time." "The tides of science and technology cannot be held back; our institutions must grow with them or be engulfed by them."

The author's second point (that institutional growth has not been coordinated) is, in a certain sense, paradoxical when considered in light of the first. In the opinion of Von Mehren, there

has recently been "a proliferation of international and regional institutions" on the international scene concerning the atom. "While it is true that many new institutions have been created, the complex problems and delicate questions of relationship, cooperation, and competition which they pose have not been solved." The institutional picture on the international level shows various avenues of international cooperation. The United States, Canada, and the United Kingdom have bilateral programs. As of Jan. 1, 1958, the American program had been implemented by 39 bilateral agreements with 37 different nations. Western Europe has two regional planning groups: (1) the European Atomic Energy Community (Euratom) and (2) the European Nuclear Energy Agency of the Organization for European Economic Cooperation (OEEC). Finally, there is the International Atomic Energy Agency. "The problem of the relationship between these various institutional approaches to the peaceful international [use of atomic energy] is difficult and complex" and does not lead to a close working relationship between the two groups. In summary, Von Mehren states that, once the policies of the various international, regional, and bilateral programs have hardened and taken definite form, it may be too late to achieve a solution.

It is the author's belief that the traditional legal problems posed are difficult and intricate but that nuclear power does not pose any unique legal problems. In general, the problems are similar to those with which the lawyer commonly deals, although the problem is often presented in difficult factual environments.

In the international field the author poses these questions: "What are the legal problems of international trade in this area? What form should American enterprise choose for entry into foreign atomic markets? How will America's foreign trade in atomic reactors be affected by our bilateral program; by our antitrust laws; by potential tort liability; by patent problems; and by problems of foreign-exchange convertibility?"

In further discourse on his subject, Von Mehren considers a serious nuclear incident occurring at a nuclear power station in Düsseldorf with persons and property in Belgium, France, Luxembourg, the Netherlands, and West Germany suffering injury. What would be the legal liability of the Americans if they had fabricated and supplied the fuel elements to this power station? The answer is not easy, he says. "It raises difficult questions of conflicts of law, the applicability of the theory of MacPherson vs. Buick Company" (217 N. Y. 382, 1916, a manufacturer is liable for latent defects in his products) in continental law systems, "and the enforceability of foreign judgments in American courts." With respect to tort liability, the author believes that the significant point is that it, like problems of atomic waste disposal, "is inherently a multinational problem" and that it, "like most legal problems in the atomic field, is not unique but is a traditional problem posed in a new environment." (David G. Schultz)

Liability and Indemnification

International problems of financial protection against nuclear liability are considered to be among the principal limiting factors on the growth of a truly international atomic industry. A study of the problems has been conducted at Harvard under the joint auspices of the Harvard Law School and the Atomic Industrial Forum, Inc., by a team headed by Robert Eichholz, former counsel to the U. S. foreign aid program, and the results of the study have been summarized in *The Forum Memo to Members*.[22] The highlights of the report are summarized below:

1. *Ideal Solution.* "Ideally," the report says, "a solution is called for which would embrace all nations participating in the development of nuclear power and which would deal with all phases of the third-party liability problem." In the opinion of the authors of the report, any such solution should meet a number of tests, including the following:

a. The operator of a nuclear facility should be liable to the public for injuries caused by that facility, "irrespective of whether he or anyone else was negligent."

b. The operator's liability should be limited to obtainable financial protection.

c. During facility operation, financial security should be maintained for the compensation of victims.

d. In the absence of some control, the supplier should not be chargeable with absolute liability if the operator has absolute liability.

e. Uniform periods of limitation should be provided for the filing of claims; two years is suggested if injury is ascertainable, ten years if it is not.

f. "Since the financial security provided by the operator will be insufficient to satisfy all judgments in full in case of a major accident, the country in which a nuclear facility is located should recognize the obligation to provide additional compensation out of public funds, up to a specified minimum aggregate amount . . . without reserving the right of recourse."

2. *Regional Solutions.* It is recognized that "a world-wide convention could not be negotiated until after many years of effort;" however, the report notes that an interim convention under the auspices of the OEEC could serve as a basis for further international cooperation. The OEEC convention limits an operator's liability to $15 million, but a signatory by a national legislation may reduce this amount to $5 million." This leaves the entire European public with grossly inadequate protection in the event of a moderately large-scale incident. The report suggests that the Euratom nations could collectively adopt additional measures beyond the OEEC convention and that "these might take the form of national indemnification legislation by each of the six nations, on the American or German model. Alternatively and perhaps preferably, each of the six might contribute to a common European indemnity pool."

3. *Other Interim Solutions.* As to what the U. S. Government might be able to do to supplement the provisions of the OEEC convention, the report suggests three principal alternatives:

a. "The Congress could enact legislation to the effect that the aggregate recovery from a defendant in American courts for foreign nuclear accidents was not to exceed the amount to which recovery was limited under the law of the country in which the accident took place."

b. The United States could become a limited participant in the regional convention by accepting it with the reservation that it applied "in the United States only with respect to accidents occurring outside its territory."

c. In the case of nuclear development programs requiring extensive U. S. assistance (such as the proposed Euratom program), the U. S. Government could seek "a hold-harmless undertaking for American suppliers, at least on a temporary basis."

The report also recommends that consideration be given to another possible measure for the protection of the American supplier. It suggests that, by analogy to the currency convertibility and expropriation guaranties available under the Mutual Security Act, U. S. Government guaranties against nuclear liability arising out of foreign nuclear incidents should be made available to American suppliers for a fee. These guaranties would be to protect the supplier "against claims which are reduced to judgment against him in American courts. The guaranty would only be for the amount of such judgment in excess of the private insurance available for him." (David G. Schultz)

The participation of the United States in discussions with other nations on liability problems has been in effect for some time.[23] U. S. efforts "have been directed primarily toward encouraging and supporting efforts of foreign governments to enact their own legislation and effect international agreements."

On the national level, tort liability arising from nuclear reactors is an important problem resulting from private and public use of atomic energy, as discussed by White[24] in a recent article. The case of American Alliance Insurance Company vs. Keleket, in which a tiny explosion occurred in a finger-tip-size capsule of radioactive radium salt and resulted in a long shutdown of the building and a $250,000 decontamination expense, is cited as typical of a large number of difficult and complex cases that may be expected.

It is the opinion of White that time factors may increase and that damages may multiply in accidents involving atomic reactors. Since remote effects may appear generations later, the author contends that the court is not the correct institution to be used, principally because of inexperience. A Federal compensation board or agency, patterned after state workmen's compensation boards, is suggested. Premiums, however, would be charged to the atomic energy or radiation users, and the Government would pay the judgments of the Federal board. In this suggestion, this reviewer finds the Government injected into a large-scale insurance business. The author fails to mention an appeals procedure; such, however, is believed to be implied since appeals of some nature are inherent in most hearings.

In summation, the author's opinions as to the increase in problems which may arise from nuclear incidents are not new, nor does the handling of cases by a Government board or agency

Table I-1 ACTIVITIES OF THE VARIOUS STATES IN ATOMIC ENERGY*

(Dates in columns 2 and 4 refer to first such action. In several states, more than one committee has been appointed and codes have been expanded or revised several times. Other dates are the dates the actions became effective.)

State	Has major code or regulations	Requires registration of sources	Has state committee or office of authorized coordinator	Allows exclusion of radiation damage from fire insurance	Includes radiation injury under workmen's compensation	Has built or started action to build own facility	Requires state approval for construction of nuclear facilities
Alabama†			‡				
Alaska		‡	June 27, 1959, §	June 27, 1959			
Arizona							
Arkansas†	June 11, 1959		‡, §				
California	‡		Sept. 20, 1959, ‡, §	Sept. 20, 1959			
Colorado		January 1959	‡				
Connecticut¶	October 1957	October 1957	‡, §	June 2, 1958			
Delaware		‡	‡				
Florida†			‡, §				
Georgia†			‡, §			August 1957	
Hawaii							
Idaho				March 1959			
Illinois		‡	July 1959				
Indiana	1959	1959	1959				
Iowa			‡		1959		
Kansas	July 1, 1959	1959	1959		1959		July 1, 1959
Kentucky†			July 1957, ‡, §				
Louisiana†			‡				
Maine			§	June 2, 1958	1959		
Maryland†			1957, ‡				
Massachusetts	December 1957	‡	‡, §	June 2, 1958			
Michigan	‡	‡	‡	1959			
Minnesota**		1959	‡	1959			Dec. 17, 1958
Mississippi†			June 17, 1958				
Missouri			1959				
Montana							

Mississippi†			June 17, 1958			
Missouri			1959			
Montana						
Nebraska			1959			
Nevada				July 1, 1959		
New Hampshire			§			
New Jersey			July 8, 1959			
New Mexico	June 12, 1959	June 12, 1959	June 12, 1959			
New York	‡	‡	‡, §			April 3, 1959
North Carolina	June 1959		June 1959			‡
North Dakota		‡		July 1, 1959		
Ohio	1957	1959	1957, ‡, §			
Oklahoma†			‡			
Oregon			August 1958	May 1959	May 1959	
Pennsylvania	‡	‡	March 1959	1959		
Rhode Island			‡, §	June 2, 1958		Nov. 4, 1958
South Carolina†			1959			
South Dakota		‡		July 1, 1959		
Tennessee†	July 1, 1959	July 1, 1959	July 1957, ‡, §		July 1, 1959	
Texas†	‡	‡	‡		1959	
Utah						
Vermont				June 2, 1958		
Virginia†			1956		Mar. 29, 1958	
Washington			March 1957, §			
West Virginia†				July 1, 1959		
Wisconsin			‡			
Wyoming		‡				

*From *The Forum Memo to Members* of the Atomic Industrial Forum, Inc., December 1957 to August 1959.
†Participant in Southern Regional Advisory Council on Nuclear Energy.
‡Action has been taken, but the date of action was not available.
§State coordinator has been named or authorized.
¶On June 13, 1950, Connecticut legislated maximum permissible radiation doses.
**The Minnesota state committee has reported that, in June 1958, the AEC authorized the state to take responsibility for some phases of inspection and control.

argue convincingly for complainants' causes with the Government paying the damages.

(W. L. Harwell)

The current AEC provisions for the indemnification of licensees and of activities conducted under contract with the Commission have evolved from Public Law 85-256. These provisions for the indemnification of licensees are temporary, and a revision is currently under consideration. Under the most recent draft policy,[23] "the total aggregate liability of the insurance syndicates is limited by the sum of the limits of liability of all the applicable policies." Also:

> Licensees subject to the financial protection requirements of Section 170 of the Act are required to furnish financial protection for public liability arising out of a nuclear incident. Above the amount of financial protection, the Commission furnishes indemnity protection in the amount of $500 million. The financial protection furnished by the licensees and the indemnity protection furnished by the Commission should meet as closely as possible in order to avoid any gaps between the two. While to a very large extent this objective has been accomplished, there remain some unresolved problems . . . .

A number of matters currently under consideration for revision are listed in the report.

The AEC report[23] also reviews the status of the maritime indemnity problem. These problems were resolved for the N.S. *Savannah* as a specified single case by the passage of Public Law 85-602. It is hoped that international agreements on nuclear ship liability and admiralty will be sufficiently clarified in the next few months to indicate what general legislation will be necessary. (W. B. Cottrell)

Even if agreements were reached as to the indemnification to be extended by the liable party, there would still remain the problem of proving damage. A recent article on this problem by Hutton[25] was directed to the attention of both the technical expert and the lawyer, with special emphasis on the important role that the technical expert can anticipate playing in future legal actions based on radiation injury. Radiation injury is not to be regarded as new, but its potential is greatly magnified today. This article discusses only briefly some of the evidentiary problems that may arise in litigation founded on radiation injury; however, it serves to alert the technical expert to the fact that he can expect to play an important part in the development of evidence on behalf of both the plaintiff and the defendant as this field of litigation develops.

Although the ordinary rules of evidence will apply, it will be necessary for the lawyer to employ the technical expert for his understanding of the technology involved and his knowledge of applicable trade language and practices to develop the proof, either pro or con, of actionable radiation injury. This will be in addition to the customary expert medical testimony to either associate an injury or disease with radiation or, on the other hand, to distinguish its other actual or presumptive cause or causes.

In this article the author discusses certain circumstances in which the technical expert will provide key evidence to determine an issue in a court of law. In addition, Hutton offers a few suggestions which he thinks will prove to be invaluable to the defendant's technical expert when the time comes for the expert to testify.

These suggestions are as follows:

1. Keep full and complete records of safety procedures and detailed accounts of any radiation incidents.
2. Maintain the records beyond the apparent statute of limitations.
3. Subscribe to a good film badge service and retain the films. Maintain records of how they were issued, identified, and worn. Be prepared to defend their accuracy.
4. Comply with pertinent regulations.
5. The best solution to a legal problem is to anticipate it and avoid it—keep in close touch with counsel.

(E. J. Glennon, Jr.)

Activities of the Various States in Atomic Energy

State legislation, regulations, and rules prior to about 1955 have been found wanting in many respects when they have been applied to the control of hazards associated with radioactivity releases. The legal problems of radiation damage have appeared and grown, beginning with the earliest recognized hazards of radium to watch-dial painters and of medical X-ray damage to patients and doctors (followed by the recognition of hazards of foot fluoroscopy* and nuclear ex-

*Most states have prohibited X-ray or fluoroscopic shoe-fitting devices, the most recent ones being Arkansas and Minnesota.

plosion fallout), to the already large and rapidly expanding field of the manufacture, shipment, and use of radioisotopes and the still more rapidly expanding problems associated with atomic power production and disposal of radioactive wastes. Studies and legislative activities by states are now rapidly increasing. A survey[26] of several phases of state activities through July 1958 is summarized in Table I-1.

The survey[26] abstracts the related activities of six Federal Government entities and seven other groups. It presents several proposed model state statutes and codes prepared by governmental and private sources. A bibliography of 43 selected references, listed chronologically from 1954 to the present time is presented as a directory of both detailed information and broad background. (L. A. Mann)

References

1. Research and Development in Reactor Safety. A Program of the United States Atomic Energy Commission, U. S. Government Printing Office, Washington 25, D. C., 1959.
2. Reactor Fuses, *Nuclear Safety*, 1(1): 17-18 (September 1959).
3. Fission-product Release, *Nuclear Safety*, 1(1): 32-33 (September 1959).
4. Structural Integrity of Containment Vessels, *Nuclear Safety*, 1(1): 26-29 (September 1959).
5. Site-selection Criteria, *Nuclear Safety*, 1(1): 4-5 (September 1959).
6. Meteorology and Atomic Energy, AECU-3066, July 1955. (Unclassified AEC report.)
7. Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants, WASH-740, March 1957. (Unclassified AEC report.)
8. W. B. Cottrell et al., Oak Ridge National Laboratory, Sept. 9, 1959. (Unpublished.)
9. Permissible Dose from External Sources of Ionizing Radiation, Addendum to National Bureau of Standards Handbook 59, U. S. Department of Commerce, Apr. 15, 1958.
10. "Radiation Protection," Pergamon Press, New York, 1958.
11. C. R. McCullough, in *Health Physics Journal* (to be published).
12. Maximum Permissible Dietary Contamination after the Accidental Release of Radioactive Material from a Nuclear Reactor, *British Medical Journal*, 1: 967-969 (1959).
13. F. R. Farmer and P. T. Fletcher, Siting in Relation to Normal Reactor Operations and Accident Conditions, presented at International Symposium on the Safety and Location of Nuclear Plants, June 16–19, 1959, Rome, Italy.
14.* H. J. Dunster et al., District Surveys Following the Windscale Incident, October 1957, A/CONF. 15/P/316.
15. *Nuclear Safety*, 1(1): 37-38 (September 1959).
16. Code of Federal Regulations, Title 10, Part 71.
17. Regulations to Protect Against Accidental Conditions of Criticality in the Shipment of Special Nuclear Material, *Atomic Energy Law Journal*, 1(1): 101-102 (1959).
18. H. F. Henry, comp., Guide to Shipment of U^{235} Enriched Uranium Materials, TID-7019, June 1959. (Unclassified AEC report.)
19. H. F. Henry et al., Criticality Data and Nuclear Safety Guide Applicable to the Oak Ridge Gaseous Diffusion Plant, K-1019 (5th Rev.), May 22, 1959. (Unclassified AEC report.)
20. H. F. Henry et al., Carbide and Carbon Chemicals Company, Oct. 5, 1951. (Unpublished.)
21. R. B. Von Mehren, The Development and Use of Nuclear Power—Some Reflections on Legal Problems, *Atomic Energy Law Journal*, 1(1): 3-22 (1959).
22. Solutions Proposed to World Liability Problem, *The Forum Memo to Members*, Atomic Industrial Forum, Inc., 6(2): 22-24 (February 1959).
23. U. S. Atomic Energy Commission, Indemnification of Atomic Energy Activities and Operations of Advisory Committee on Reactor Safeguards, 1958-1959 Report to the Joint Committee on Atomic Energy, p. 8, Mar. 31, 1959.
24. W. L. White, Jr., Problems of Tort Liability, Arising from Nuclear Reactors, *Atomic Energy Law Journal*, 1(2): 164-174 (1959).
25. G. L. Hutton, Evidentiary Problems in Proving Radiation Injury, *American Industrial Hygiene Association Quarterly*, 20(2): 111-112 (April 1959).
26. J. S. Harris, ed., State Activities in Atomic Energy, A Forum Survey, Atomic Industrial Forum, Inc., News Release, July 10, 1958.

*Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1958. (Individual papers are available from the Office of Technical Services, Department of Commerce, Washington 25, D. C. Bound volumes of papers are available from United Nations, Office of Public Information, New York.)

Fission-product Release from UO_2

Uranium dioxide is a promising fuel material for power reactors because of its radiation stability at high temperatures; however, certain potential hazards arise from its use. Although fission products are accommodated better within the UO_2 lattice than within metallic fuel materials, significant amounts of fission products are released by a diffusion mechanism at the temperatures that will exist in the fuel of an operating power reactor. The fission products that are released create a potential hazard as a result of the pressure build-up within the metal container surrounding the UO_2 material, and they also represent a source of activity release in the event of a defect or rupture of the metal container. A number of reports discussed in the previous issue of this Review[1] dealt with the release of fission products in a fuel-element meltdown. The reports reviewed here are concerned with the long-term release of fission products from UO_2 fuels at operating temperatures.

The release of fission gas from UO_2 occurs by the process of solid-state diffusion, and the quantity of fission products released may be estimated by the methods described[2] in Report WAPD-183 and by Booth.[3] The calculation of the quantity of fission-product release from UO_2 depends on knowledge of the temperature dependence of the diffusion coefficient, D, or the diffusing nuclei in the UO_2, the diffusion path, a, or the diffusion parameter, D' (where $D' = D/a^2$). It is also necessary to know the temperature distribution in the fuel for the conditions of operation of the reactor.

The temperature distribution in the fuel material during reactor operation is difficult to ascertain because instruments capable of accurately measuring high temperatures in a radiation field are not yet sufficiently developed. Therefore, temperature distributions in UO_2 fuel elements have been estimated by two methods, each having a high degree of uncertainty. Values of the heating rate in the fuel, the cladding surface temperature, the fuel-element geometry, and an assumed value of the thermal conductivity of UO_2 are used in one method. This method introduces two uncertainties because the thermal-conductivity values reported by various investigators for irradiated and unirradiated UO_2 are not in agreement and because no allowance is made for the existence of a gap between the UO_2 and the metal container or for discontinuities in the temperature distribution due to cracking of the UO_2. The second method for estimating the temperature distribution in the fuel is based on observations of grain growth and/or melting of the irradiated fuel.

In a recent review of the literature pertaining to UO_2 fuel elements, Runnalls[4] reports that temperature distributions in a cylindrical fuel element may be conveniently characterized by the following expression:

$$\int_{T_s}^{T_r} k(\theta)\, d\theta = \frac{HR}{2} \text{ fn } (\kappa b, r)$$

where

$$\text{fn } (\kappa b, r) = \frac{2I_0(\kappa b) - I_0(\kappa r)}{(\kappa b)\ I_1(\kappa r)}$$

which has the value unity at $r = 0$ flux for negligible depression and, hence, uniform heat generation, and where

$k(\theta)$ = thermal conductivity of UO_2 at temperature
T_r = temperature at radius r
T_s = temperature at UO_2 surface
H = sheath-coolant-surface heat flux
R = radius of outside of sheath
b = radius of UO_2
I_0 = modified Bessel function of zero order
I_1 = modified Bessel function of first order
κ = reciprocal of effective diffusion length in UO_2

Robertson et al.[5] have shown that this expression is a useful one since it may be applied without the precise knowledge of either the thermal conductivity or the axial temperatures in the

fuel. A curve of $\int_0^T k(\theta)\, d\theta$ vs. T is given by Runnalls which is based on data from experiments at Chalk River and at Westinghouse.[4] In arriving at some of the points on this curve, it was assumed that discernible grain growth would be produced in ADU type UO_2 at a temperature of 1500°C. The estimation of the temperature for grain growth is essential in determining the shape of this curve. Although the various tests were performed on materials from different vendors, it was assumed that the activities of the powders were such as to result in grain growth at 1500°C. Since little information exists as to the effect of irradiation on the time-temperature relation of grain-growth rates, there is considerable uncertainty associated with estimating temperatures by observation of the grain growth.

Numerous experiments have been made in an attempt to determine the diffusion rates of xenon and krypton from UO_2. Cottrell et al.[6] have reported that both Westinghouse and Chalk River are now determining the diffusion parameter by out-of-pile measurements because this type of experiment permits an accurate knowledge of the temperature. They also point out that "Data presently available in the published literature on diffusion coefficients for gases from UO_2 are not adequate to predict the release of fission products from uranium oxide. The values presently given for D and D' are suspect." Some recent unpublished data of Auskern[7] concerning the diffusion coefficients of Kr^{85} in UO_2 powders are presented in Table II-1.

Table II-1 DIFFUSION COEFFICIENT FOR Kr^{85} IN UO_2 POWDERS

Type of UO_2 powder	Temp. range, °C	Diffusion coefficient, cm^2/sec	Surface area, m^2/g
NUR	900–1400	$4.9 \times 10^{-4}\, e^{-73,800/RT}$	1.3
Crushed PWR pellets	900–1500	$2.6 \times 10^{-4}\, e^{-65,600/RT}$	0.1

Runnalls[4] points out in his paper that the factors influencing gas release and the actual escape mechanism are still not clearly understood. He cites the work of Booth and Rymer,[8] who report that markedly varying amounts of gas were evolved from samples of similar density prepared by different fabrication methods. Even for specimens which were prepared for virtually the same fabrication techniques in the laboratory and which received similar irradiation, the fission-gas release varied appreciably.

The problem of pressure build-up in metal-clad UO_2 fuel elements will vary in degree, depending on the conditions of operation of the reactor and the design of the fuel element. Lustman[9] has shown that, even with the most pessimistic assumptions, fission-gas release will not limit the performance of the present UO_2 core in the Shippingport reactor. However, Runnalls[4] points out that, in reactors operating with lower pressures external to the fuel and with larger ratios of fuel diameter to sheath thickness, the build-up of internal gas pressure might prove to be a severe limitation. Cottrell et al.[6] point out that most of the studies pertaining to UO_2 have been for the conditions that exist in water-cooled reactors operating with low cladding temperatures and high external pressures. Conditions in a gas-cooled reactor may create additional significant problems because of low system pressures and higher cladding temperatures. If the fuel-cladding temperatures are high enough in gas-cooled reactors, cesium may contribute significantly to the pressure build-up normally attributed only to xenon and krypton. The initiation of additional work at Battelle Memorial Institute to determine diffusion coefficients of fission products in UO_2 has been reported.[10]

(J. L. Scott and H. N. Culver)

Chemical Reactions of Reactor Materials

The chemical reactions involving materials that are of concern to the reactor designer can be conveniently divided into two classes: (1) those reactions which take place at very slow rates over long periods of time during normal operation of the reactor and which will have a long-range influence on reactor lifetime, design specifications, and economics and (2) those reactions which can take place rapidly and could cause, or be important in, a reactor incident. With a few exceptions, most of the studies reviewed here are concerned with reactions which can cause reactor incidents or which may take place as secondary phenomena during a reactor accident and thus magnify the severity of the event. These considerations are therefore an important part of reactor safety. As a general

rule, the reactions covered under the first classification are the ones most likely to affect reactor design and economics. Some recent studies pertinent to the use of pyrophoric metals and organic coolants and a possible explosion hazard are discussed here.

Reactions of Reactor Metals

The study of the reactions of pyrophoric metals with both air and water is receiving much attention at several installations. Two of the studies discussed here are concerned with the long-term behavior of these materials under normal reactor operating conditions. The remainder concern the rapid reactions of uranium, zirconium, and titanium with air (oxygen) and water. These are important to the reactor designer since they may cause, or greatly magnify the severity of, a reactor accident.

The complexity and difficulty of the problems involved in the use of pyrophoric metals have led to many different types of experiments. In turn, the different experimental conditions have made a comparison of the results difficult, and there is frequently uncertainty as to the applicability of the results of the experiments to conditions significantly different from those prevailing in the laboratory. This will be apparent in the results of some of the ignition experiments discussed below.

1. *Oxidation Reactions.* The Argonne National Laboratory (ANL) has reported results of studies of the effect of the specific area of the sample on the ignition temperature* of uranium and certain uranium alloys in oxygen.[11,12] These data were obtained by the so-called "burning-curve" method.† Similar experiments in air were unsuccessful because of the tendency of samples to thermal cycle rather than to ignite. In a modified experiment where oxygen (or air) is suddenly admitted to the furnace tube in which the sample is at an equilibrium temperature (and the metal either ignites or it does not), good results were obtained with oxygen, and the ignition temperatures were determined quite accurately. Once again, however, the experiments in air gave uncertain results, and the sample showed a tendency to thermal cycle.[13] This behavior is substantiated by unpublished data obtained at Hanford.[14]

Burning-curve experiments were also carried out in oxygen-helium and oxygen-nitrogen atmospheres.[13] These experiments have yielded both ignition temperatures and maximum burning temperatures. Some evidence for thermal cycling was noted even in pure oxygen, where the maximum temperature was over 2000°C.

The maximum burning temperatures and the ignition temperatures vs. the oxygen concentration in the gas mixture were also obtained. The report[13] states that, as the maximum burning temperature decreases, the pyrotechnic effects are more subdued; and, where the ignition temperature curve joins the burning temperature curve, no ignition is observable.

The rates of propagation of the oxidation front and the ignition front in uranium and uranium alloys were also studied at ANL. In air, the rates of propagation of the two fronts are equal to within an experimental error of ± 5 per cent. In oxygen, the ignition front is propagated about 10 times as fast as that in air, but no oxidation front is observable because of the light intensity and the faster rate of propagation of the ignition front. The rates of propagation of burning (in centimeters per second) for uranium samples 3 mm wide and 6 cm long are given by the equations

$$\text{Rate} = \frac{0.137}{B^{0.696}} \quad \text{in air} \tag{1}$$

$$\text{Rate} = \frac{0.890}{B^{0.801}} \quad \text{in oxygen} \tag{2}$$

where B is the sample thickness in millimeters. The higher propagation rate in oxygen is attributed to the much higher burning temperature.

The ignition and oxidation of zirconium have been studied at ANL[13] and at the Stanford Research Institute.[15] The different methods employed emphasize the difficulty of comparing the results of the two studies.

*The ignition temperature can probably be best defined as the temperature above which a sample of given dimensions in a specified atmosphere will spontaneously ignite and be heated sufficiently by the exothermic reaction so that it will continue to be burned until entirely consumed.

†In the burning-curve method, the temperature of the sample is continuously recorded as a function of time. The ignition temperature is characterized by a sharp temperature spike in the curve which occurs when the sample, slowly heated by oxidation, reaches the ignition temperature.

The Stanford experiments[15] were conducted in a heated stainless-steel pressure vessel equipped with a hydraulic pull-ram that was used to rupture the sample and expose a fresh surface. The samples were held at a constant temperature under a given oxygen pressure and then broken. The experiment was then repeated at another oxygen pressure. At a fixed temperature the minimum oxygen pressure at which ignition would occur was taken as the lower ignition limit. In these experiments, the reproducibility with zirconium was poor, possibly because of nonuniform surfaces of the specimens. (Some specimens ignited, but propagation depended on the sample cross section.) A modified experiment in which the specimen was annealed in a vacuum before being exposed to the oxygen produced more reliable data. Test strips of zirconium ($0.5 \times 3 \times 0.020$ in.) ignited spontaneously at 250 to 300 psi of oxygen at room temperature.

Shielded-ignition experiments on zirconium and zirconium alloys at ANL[13] have shown that (1) the ignition temperatures in oxygen and in air are the same within experimental error; (2) the ignition temperatures in oxygen are not dependent on the flow rate of the oxygen, whereas the ignition temperatures in air are lowered as much as 20°C by increasing the flow rate; and (3) the ignition temperature for zirconium is dependent on the specific area of the metal. The ignition temperature of zirconium foils (T_{ig}, in degrees Centigrade) is given by the following equation:

$$T_{ig} = 1070 - 208 \log (A/m)$$

where A/m is the specific area of the sample in square centimeters per gram.

Spontaneous ignition studies on titanium tensile specimens at Stanford Research Institute[15] gave more reproducible data than did the studies on zirconium. It was noted that, even if the sample ignited, the propagation of the reaction through the entire sample depended on the sample cross section. The oxygen-pressure ignition limits were 350 psi at room temperature, 210 psi at 300°C, 150 psi at 500°C, and 75 psi at 1000°C.

The difficulty of comparing the Argonne and Stanford results that were obtained under different conditions is apparent. It should probably be emphasized that the ignition temperatures are more a parameter of the conditions of the experiment than an intrinsic property of the metal since such factors as surface roughness and oxide coating may have a significant effect.

Low-temperature isothermal studies of the oxidation of uranium have also been reported by ANL.[13] These studies have included microscopic observation of the oxide layer formed by the oxidation of uranium at 200°C and 200-mm-Hg oxygen pressure. The application of a Tesla coil to the reaction tube during several runs showed a marked increase in the oxidation rate. Similar effects have been noted by other workers, and this phenomenon is being investigated further at several sites since, for some materials at least,[16] the presence of ionizing radiation in a reactor increases the oxidation in comparison with that observed out-of-pile.

A detailed discussion of the isothermal oxidation data for zirconium and zirconium alloys at 700°C and 200-mm-Hg oxygen pressure, reported previously by ANL,[11] is given in a recent report.[13] The data are compared with the predictions of the Wagner-Hauffe theory, and the effect of alloying additions to the zirconium on the oxidation rate is predicted. In most cases the data are in agreement with the theory. The addition of 1 to 4 at.% titanium to zirconium greatly increased the oxidation rate of the zirconium.[13]

2. *Metal-Water Reactions.* The wide use of water as a coolant and moderator for solid-fuel reactors makes the knowledge of the kinetics of metal-water reactions imperative before such reactors are operated at power densities that can lead to temperatures at which these reactions will take place. Argonne[17] has reported some preliminary data obtained in an experimental program directed toward determination of the rates of reaction of molten reactor fuels and cladding metals with water.

Hopkinson[18] has published a study of the kinetics of the uranium-steam reaction. The data were obtained by continuously recording the change of weight of the uranium specimen on a modified thermal balance. The experiments covered the temperature range from 160 to 1400°C.

The measured reaction rates showed some scatter. This was attributed to (1) self-heating of the sample because of the exothermic reaction and (2) the possibility of oxygen impurity in the steam supply. The scatter of the data might have been improved if it had been possible

to measure the sample temperature during the run rather than assuming it to be the same as that of the gas stream. This is probably a serious source of error.

The reaction was reported to follow a linear rate law from 160 to 880°C. From 880 to 1060°C, the initial rate followed a parabolic law, but it changed to a linear law in 60 to 120 min. The different rates are attributed to the changing character of the oxide film as a function of temperature. Above 880°C, the oxide formed an adherent shell around the specimen which remained intact on cooling. Below 880°C, the oxide film was continuously shed by the specimen, and a fresh uranium surface was exposed. This reasoning is substantiated by the observation that, once the scale adheres to the specimen at temperatures above 880°C, the linear oxidation rate is lower than the rate at temperatures below 880°C.

The effects of reactions between zirconium and uranium-niobium alloys with water subsequent to a loss-of-coolant accident in a reactor have been studied at the Bettis Plant.[19] The calculations indicate that the exothermic heat of reaction can add significantly to the reactor decay heat and cause melting of the reactor core. It is shown that, at lower temperatures at least, a nominal amount of cooling will prevent further heat-up and melting of the core.

The reactions of sodium metal with water and steam (including D_2O and D_2O vapor) have been studied by the Nuclear Development Corporation of America.[20] The rate of reaction was followed by measuring the rate of hydrogen evolution, either manually or with an automatic recording gasometer for fast reactions.

The rate of reaction of sodium droplets with liquid water was reported to be proportional to the area of the droplet surface in contact with the water, at least up to moderate reaction rates. Cool molten sodium reacted more rapidly with liquid D_2O than with H_2O. This was attributed to the higher viscosity of the D_2O which reduced the cooling action of the water on the sodium. The reaction of hot sodium with liquid H_2O was more rapid than with liquid D_2O. The reason for this different behavior is obscure.

The temperatures of the sodium in the vapor-phase reactions were measured by placing a thermocouple in the pool of molten sodium in a tube furnace and passing the water (or D_2O) vapor over the sodium. The reaction rates for H_2O and D_2O vapors were of the same order of magnitude to within experimental error.

Organic Reactor Materials

Organic compounds have been proposed for nuclear reactor coolant so that the reactors may be operated at lower pressures than aqueous systems operated at the same temperature. Furthermore, the use of organic compounds virtually eliminates corrosion problems.

To be useful as coolants and moderators in nuclear reactors, organic materials must have (1) thermal stability, (2) radiation stability, (3) low melting point, (4) high boiling point, and (5) low cross section. Usually some compromises must be made with these requirements, and less-than-ideal materials are accepted as possibly useful. The physical properties of the organic coolants seriously considered for reactor use are cited in reference 21.

The hazards associated with the operation of nuclear reactors moderated and cooled by organic liquids arise from two sources: (1) the effect of radiolytic or pyrolytic decomposition of the organic molecules on the coolant properties and (2) the combustibility of the organic liquids. These topics are treated separately in the following section.

1. *Coolant Decomposition.* The radiolytic decomposition of the polynuclear aromatic molecules of the coolant results almost entirely in the production of high-boiling compounds of about twice the molecular weight of the original coolant.[22] Although the decomposition rates[23] reported early in the operation of the Organic Moderated Reactor Experiment (OMRE) were quite high and led to predictions of a high rate of coolant replacement, these early figures have proved to be pessimistic.[24] As might be expected, the rate of radiolytic decomposition decreases as the concentration of decomposition products in the coolant increases.[22]

The reported experience with the OMRE indicates that the presence of decomposition products in the coolant does not adversely affect the properties of the coolant or the operation of the reactor. Presently, reactors are being designed with the assumption of a steady-state concentration of 30 to 40 wt.% high-boiling compounds in the reactor. During reactor operation, it is reported that the concentration of decomposition products is kept at the above level by low-pressure distillation of the circulating coolant.[22]

2. *Combustion Hazard.* The combustion hazards associated with the use of organic coolants have been the subject of a recent study at Hanford.[21] The tests were designed to simulate the hazards associated with (1) the storage or spills of the organic compounds (the so-called "open-pan" tests) and (2) leaks of the organic coolant from the high-temperature high-pressure reactor system.

The results of the open-pan tests, which simulated coolant storage conditions or spillage, are best summarized by quoting from the report.

> 1. The cold organic liquids tested were difficult to ignite and had to be heated almost to their boiling points before ignition was easily attained.
> 2. The solid organic compounds tested, with the exception of Santowax-R, would not ignite until a portion of the compound melted.
> 3. The organic compounds burned in a sluggish manner, but they produced an intense heat.
> 4. Water was the best extinguisher for open organic-compound fires.

A leak in an organic-moderated reactor system was simulated by permitting the hot vapors to escape through a valve from a tank maintained at 350°C and pressurized at 350 psi. The results of these tests were:[21]

> 1. The escaping vapor was very easy to ignite and correspondingly easy to extinguish.
> 2. All types of extinguishers were effective against this type of fire.
> 3. Additional hazards from this type of fire are possible toxic aftereffects from exposure to the vapor and lack of visibility due to the billowing vapor cloud.

The general conclusions of the report are that a minimum combustion hazard exists in storage or make-up areas of the reactor installation.

Some hazard exists when the hot organic-coolant vapors escape from the reactor piping. This hazard is somewhat minimized by the fact that the vapors are easily extinguished if they should ignite.

Both situations can be adequately handled, however, by protecting the areas with remotely operated water-spray systems, although adequate provision must exist to prevent the core from being flooded with water. Therefore, it appears that the principal concern of the reactor designer considering an organic-moderated reactor for a power-producing station will be the economic problem of replacing the coolant destroyed by the radiation. This cost has recently been estimated at 0.73 and 0.62 mill/kw(e)-hr for 150- and 300-Mw(e) plants, respectively. This estimate[22] is based on a make-up rate of 27 lb of coolant per megawatt day per ton and a coolant cost of $0.17 per pound.

Ozone

An Oak Ridge report[25] detailing several incidents involving explosions in cryogenic apparatus operated in radiation fields should be mentioned since a gaseous waste-disposal system, employing a liquid-nitrogen-cooled charcoal trap to remove gases generated in the primary system will be employed on the N.S. *Savannah*.[26] It is well known that liquid-nitrogen traps exposed to the atmosphere will tend to condense oxygen over a period of time. If the liquid level in the trap is kept constant by merely replacing the evaporated liquid, the oxygen concentration in the trap increases. Many explosions due to the presence of liquid oxygen in these traps have occurred in laboratories, and, since the problem is now well recognized, appropriate safeguards can be instituted. In a radiation field, however, the problem is greatly magnified by the formation of ozone, whose explosive tendencies are much greater than those of oxygen. The phenomena of ozone explosions are also less well understood. The ozone further tends to concentrate in the apparatus because of its even higher boiling point.

Ozone explosions can arise from either (1) the presence of readily oxidizable materials (particularly organic compounds) or (2) the explosive rearrangement of the ozone molecule to form molecular oxygen (a lower energy state).

The hazards associated with the use of cryogenic equipment in radiation fields may be minimized by observing the following precautions:[25] (1) operating all low-temperature surfaces in a closed system such that liquid oxygen (or ozone) from the air cannot condense in the system and (2) limiting the radiation exposure (all known incidents to date have followed exposures of greater than 10^7 r). Specifically, the low-temperature off-gas system of the N.S. *Savannah* is not considered a hazard because the liquid-nitrogen refrigeration loop is a closed system and the off-gas (principally hydrogen) passes through a recombiner before it reaches the cold trap. (Sidney Langer)

Release of Wigner Energy from Graphite

The reactor irradiation of graphite produces carbon-atom dislocations that leave the graphite in a higher energy state. The increase in vibrational energy of the atoms on heating causes these carbon atoms to become relocated and, at the same time, releases the energy associated with the dislocation. This energy appears as an increase in the graphite temperature. This phenomenon, first recognized by Wigner, is of major importance in the operation of reactors containing graphite.

Several papers on Wigner energy release,[27–36] almost all of which are by Western European authors, are reviewed here. These studies are all concerned with graphite at temperatures less than 400°C, and it is evident that a great amount of study of this problem is in progress, especially in the United Kingdom. This is in contrast to the research in the United States, where the primary interest is not in reactor operation in the low-temperature range at which the Wigner effect is most significant, although the United States does have operating graphite-moderated reactors, namely, the ORNL Graphite Reactor and the Brookhaven National Laboratory Graphite Reactor, which operate with graphite at temperatures well below 400°C.

A theoretical treatment of the stored-energy annealing process in irradiated graphite, as well as an examination of the theory under reactor operating conditions with preset boundaries, has been presented by Dalmasso and Nardelli.[27] In their report the authors include some predictions on stored-energy growth and spectral distribution in graphite reactors operating at temperatures greater than 250°C.

The theoretical study makes full use of current knowledge of solid-state and lattice imperfections. It draws heavily on radiation-damage theory and experimental data and discusses a phenomenological theory of annealing by processes possessing a wide distribution of activation energies. As the authors point out, this approach has been extensively applied, and there are no substantial areas of disagreement.

In an application of the theory, the kinetics of stored-energy release in a typical graphite-moderated reactor are examined. In this study, actual experiences with the release of stored energy are reviewed in detail, particularly the first annealing program at the British Experimental Pile O (BEPO), in which reactor heating was employed. A comprehensive review of most of the unclassified literature on stored-energy annealing is also given, with special emphasis on stored-energy accumulation in graphite irradiated at temperatures above 250°C.

Although it is not stated implicitly, this study was apparently aimed at the possible prediction of stored-energy accumulation in gas-cooled graphite-moderated reactors either planned or under construction in Italy, some of which are of the Calder Hall type. The available data for very high exposures are limited, and therefore extrapolation of the data was necessary for long-term estimates. It was predicted that graphite irradiated at 300°C for 8 to 10 years in a flux of 10^{13} neutrons/cm^2 would accumulate sufficient stored energy for the specific heat curve to be exceeded at some temperature above 300°C.

This report by Dalmasso and Nardelli[27] is an excellent review of the more recent work on graphite stored energy in regard to theory, experiment, and reactor practice. The work is scholarly throughout, and, although assumptions had to be employed in the estimation of stored-energy accumulation, the assumptions are reasonable.

Laboratory Experiments on Stored Energy

An excellent instrument for adiabatic rise measurements of stored energy in graphite has been devised by Henson and Simmons. Such measurements are of great importance because they indicate the most serious conditions that can arise due to stored-energy release. The report by Henson and Simmons[28] gives details of a method for measuring stored energy in irradiated graphite in an adiabatic rise calorimeter. Since the kinetics of the energy release are not well known, the method includes an approximation and final measurement of the heat-transfer conditions so that a rate process can be described.

By one means of operation the adiabatic rise of irradiated samples is measured with considerable accuracy. In this method the sample is placed in an enclosure in which the temperature is always maintained very close to the temperature of the sample. This is accomplished through good thermal insulation between the sample and the calorimeter and an elaborate temperature-control circuit that reduces ther-

mal lagging to a minimum. The temperature of the sample is raised as quickly as possible to some given temperature where adiabatic rise conditions prevail, and the heat supply is then removed so that the subsequent temperature rise of the sample and the calorimeter is due to the release of stored energy. The maximum rise in temperature, together with the known specific heat of graphite, permits the calculation of the amount of stored energy released. It is pointed out that the greatest uncertainty in the method lies in achieving the adiabatic starting conditions instantaneously with no heat transferred immediately following the start-up. It is estimated that this uncertainty would only be 1 cal/g in the measurement of the stored energy released.

The adiabatic rise method of analysis is particularly valuable in determining the maximum temperatures to be expected in graphite reactors if a spontaneous stored-energy release should be triggered under adiabatic conditions. Although these conditions are not probable, the analysis furnishes a guide for the most serious situation that could conceivably arise.

From the accuracy of the data obtained, it is apparent that the calorimeter has the proper heat-transfer characteristics for sensitive control and that the electronic control system is sufficiently balanced to supply nearly instantaneous correction without overcontrolling. Where the purpose of the measurement is the determination of the adiabatic rise in specimens containing stored energy in excess of the normal specific heat, it is evident that the apparatus and method described in this report are equal to any previously reported.

Two other methods of analysis employed prior to the second BEPO energy release in 1958 were mentioned by Henson and Simmons.[28] One was a pseudolinear rise method, which was employed to study samples having small amounts of stored energy (presumably none in excess of the normal specific heat) and also to determine stored energies at temperatures in excess of the maximum adiabatic rise temperature. The pseudolinear rise method involves two separate runs on each sample and measurements of the rate of temperature rise on each run. In the other type of analysis, heat-transfer conditions were adjusted to simulate an actual reactor condition in which the graphite is heated or cooled by gas flow. This method was to be used in the second BEPO energy release, and several experiments were conducted to simulate reactor conditions prior to the actual reactor anneal.

In March 1958, following the second stored-energy release from the graphite in BEPO, samples were removed for postannealing measurements. The results of the measurements,[29] which are preliminary in nature, were to be used to evaluate the effectiveness of the anneal, as well as to provide information for following the subsequent stored-energy growth in the graphite.

The measurements were made using the linear rise method to determine the total amount of energy, S_0, released up to 400°C. These results were compared with specific heat data for unirradiated graphite to obtain a measure of the importance of the stored-energy effects. None of the measurements taken suggested that significant amounts of stored energy remained following the March 1958 anneal.

Previous experience had shown that an annealing temperature of 200°C would be sufficient for an adequate anneal. Temperature data taken during the anneal indicated that a temperature in excess of 200°C was reached at all points within a 15-ft-diameter cylinder, whose axis extended 3 to 15 ft from the inlet face of the reflector. Particular care was taken to explore the stored energy in the annulus, extending 2.5 ft from the radial reflector boundary. The stored energy in the annulus was low; however, even before the anneal, it was noted that the stored energy in this region was not excessively high. The graphite in the stack was sampled to discover any lack in symmetry in the anneal that would be contrary to the temperature symmetry observed. No lack of symmetry was observed, but the possibility still remained that there were significant amounts of stored energy remaining in some local annular regions where annealing was doubtful. Only two channels appeared to remain unaffected by the anneal; but, since they are adjacent to a region in which the temperature reached 200°C during the anneal, they were probably annealed.

Samples taken in December 1958, nine months after the anneal, showed a marked growth of stored energy in the central core region, about 12 ft from the inlet face of the reflector. Stored-energy accumulation in the peripheral core region was negligible. It was concluded that no appreciable amounts of stored energy existed anywhere in the reactor but that periodic checks would be advisable in the peripheral region

where annealing was doubtful, as well as any other areas where the operating temperature is low.

A brief announcement[30] in *The Forum Memo to Members* of the Atomic Industrial Forum states that Great Britain's Nuclear Power Plant Company made laboratory tests on a method dealing with the control of stored energy in graphite in the 200-Mw(e) gas-cooled graphite-moderated reactor at Latina, Italy. The technique was not described in detail, but it was said to involve localized heating of sections of the moderator by inserting electrical heaters into specified fuel-assembly casings. The method was said to be especially applicable to the bottom end of the Calder Hall type of reactor.

Annealing Procedures for Various Graphite Reactors

A recent popular article[31] (directed largely to the lay person and, to a lesser extent, to technical personnel unfamiliar with stored energy in graphite) primarily discusses the problems of graphite stored energy in the Windscale and Calder Hall reactors with respect to past experience and future expectations. In this connection the article frequently refers to the Fleck Report and draws conclusions from that report rather freely. (The author does not give a specific reference to the Fleck Report.)

In an attempt to give the reader an understanding of the causes of graphite stored energy, the author gives estimates for the lifetimes of radiation-induced defects in graphite as a function of temperature. Unfortunately, the method by which these estimates were made is not given, and therefore they are of no technical value. Drawing largely from the so-called Fleck Report, the author describes the prevailing situation in the Windscale reactors, where the amount of releasable stored energy in Windscale reactor 2 is great enough to raise the reactor temperature to 400 to 500°C if the release is accomplished by the previously employed annealing method. It is pointed out that these are dangerously high temperatures in an air-cooled reactor both for graphite oxidation and for possible uranium oxidation and subsequent release of fission products. (The Windscale reactors are now shut down.)

Referring further to the Fleck Report, it is noted that about 300 thermocouples to measure fuel temperatures, as well as 300 additional thermocouples to measure graphite temperatures, would be needed in any future annealing of the Windscale reactors. A method of releasing the stored energy in air-cooled reactors, consisting of injection heating in which hot air would heat the graphite slowly to 120°C and maintain it at that temperature for 12 hr, is also given. (These values are identical with those given in reference 30, although no source reference is given in the article.) The reactor would then be cooled to room temperature and again injection heated rapidly to 160°C, at which temperature a second portion of the low-temperature stored energy would be removed with no danger of serious temperature excursions.

It is noted that the Calder Hall reactor is in a much safer condition than the Windscale reactors, not only because its graphite temperatures are considerably higher but also because the coolant gas is CO_2, which circulates in a closed system.

There are serious failings in this article, even if it is not intended to be of a truly technical nature. It is doubtful whether the average reader knows the details of the stored-energy distribution in the Windscale reactors or that fission heat was used in previous annealing operations. The significance of these facts is that the damaged region of the pile is near the coolant-inlet face, and therefore the normal temperature distribution is unfavorable for using fission heat in the annealing operation. It is, of course, evident that injection heating would overcome this difficulty; however, it is not clear that this is either technically or economically sound.

Rimmer considered a procedure for releasing the stored energy in the graphite in Windscale reactor 2. As outlined, no fission heat would be used; instead, heated air would be passed down the channels in a controlled program of temperature elevation. The proposed annealing method would include raising the graphite temperature very slowly to 120°C through a stepped scheme totaling more than 100 hr. This would be followed by a "cooking" period at 120°C for a further 12-hr period and then cooling to room temperature. Finally, the graphite temperature would be raised exponentially to 180°C with a time constant of 1 hr, and, presumably, energy release would be triggered at about 160°C. Under these prescribed

annealing conditions and with the best knowledge of the stored-energy status in Windscale reactor 2, calculations were made to determine the maximum temperatures that would be reached during this annealing cycle.

To make this calculation, some assumptions concerning the annealing kinetics in the stored-energy release had to be made which were known to be incorrect. For example, it was necessarily assumed that all the processes releasing energy have an activation energy of 1.7 ev, whereas it is well known that these processes have a complete spectrum of activation energies. For some guidance to be available concerning the probable course of the proposed annealing, it was rightfully decided that the calculation would nevertheless be useful.

The calculation considers one-dimensional heat transfer down a single channel since it was previously shown[33] that the channels in the reactor may be considered as being independent in this type of problem with the heat-transfer coefficients and heat-flow data known. There remained a determination of the rate of stored-energy release in all parts of the reactor before the complete calculation could be worked out for high-speed computation. Sets of linear rise type curves were used in evaluating the rates of stored-energy releases, and interpolation was used where necessary to set up the computation for 2-ft intervals in the channels over a 20-ft span.

As a result of these calculations, it was estimated that the maximum graphite temperatures that would be reached under these conditions would be about 311°C. Presumably, this indicates that the low-temperature anneal (120°C) would liberate sufficient energy so that the second triggered release (at 160°C) would not result in an excessive temperature rise. Although it is not explicitly stated in the article, it appears that the objective was to achieve, within narrow limits, a 300°C anneal. This is in contrast, for example, to the approach anticipated for the graphite energy release of the ORNL Graphite Reactor,[34] as discussed below.

The ORNL Graphite Reactor (OGR), in continuous operation since November 1943, is the oldest operating reactor in the world. Since the stored energy in the graphite moderator of this structure has never been released, it also follows that this is the oldest unannealed graphite reactor in the world. Preparations are being made for an anneal, and a safeguard report[34] has been issued. This report contains a fairly complete description of the reactor, its controls and instrumentation, and its long operational history; an evaluation of the stored-energy condition of the graphite moderator; and a complete description of the method selected for the proposed annealing operation, together with an analysis of less desirable methods.

A comprehensive survey of the graphite-stored-energy condition was obtained by independent calorimetric and X-ray analyses of samples removed from the graphite moderator through remote coring methods. In making the calorimetric measurements, emphasis was placed on the temperatures (approximately 200°C) at which the stored-energy-release spectrum poses the most serious problem; in addition, measurements were made of the stored energy releasable at all temperatures up to the combustion temperature of carbon. X-ray-diffraction measurements of c_0 lattice spacings of fuel-channel graphite scrapings were correlated with the calorimetric measurements, although it is pointed out that the correlation was possible only because the reactor had never been annealed.

From graphite-moderator samples containing the greatest amount of low-temperature (200°C) stored energy, annealing data were obtained which indicated that "soaking" this graphite for several hours in the low-temperature range of 138.5 to 165°C results in diminishing that portion of the stored-energy-release spectrum which is spontaneously releasable to the level at which spontaneous adiabatic release is no longer possible. Significantly, this is accomplished with no temperature excursions through a process of slow heat evolution under ordinary heat-transfer conditions. It was estimated that only about 4 per cent of the moderator graphite contained spontaneously releasable stored energy and that this damaged region was located 4 to 12 ft from the inlet face of the reactor in the form of a dish on its edge. Thus the stored-energy region conforms to the temperature-flux distribution. Measurements of graphite-moderator growth showed a slight distortion (maximum of 0.28 in.) in this same region of maximum stored energy. This slight distortion is not considered to be detrimental to the safety of the reactor. The relatively small amount of damage to the moderator over this long period of operation is satisfactorily explained by the relatively high operating temperatures, especially

during the summer months. It is pointed out[33] that the accumulation of stored energy is strongly dependent on temperature, especially below 100°C, and that temperature variation in the 30 to 100°C range not only causes a wide variation in stored-energy accumulation but also causes a wide variation in its spectral distribution.

In consideration of the short-term and long-term stored-energy situation in the moderator, several methods were investigated as potential annealing devices. The method finally selected and proposed for approval is described as being ideally compatible with concepts of safety, effectiveness, efficiency, and cost. The method involves alteration of the present one-pass coolant system to permit the reversal of air flow through the fuel channels under conditions similar to normal reactor operation. The reversal of the air flow will reverse the temperature distribution of the moderator so that the maximum graphite temperatures will be in the regions of maximum stored energy. It is proposed that the annealing schedule proceed slowly to accomplish a low-temperature anneal that will succeed in releasing the spontaneously releasable stored energy. Subsequently, higher temperatures will be employed as needed. Since the installation of the cross-duct for the air-flow reversal will be a permanent installation, it is suggested that further anneals at shorter, infrequent intervals be used to retard the further growth of graphite-stored energy.

For several years, a reversed air-flow technique for graphite-moderated reactor annealing has been successfully employed at Brookhaven National Laboratory,[35] where the problem of moderator distortion has been paramount. Annealing temperatures have exceeded 350°C. It appears that the proposed method for annealing the ORNL Graphite Reactor is unlike any previously employed method, in the sense that low temperatures are to be employed to reduce and retard stored-energy growth under nearly normal reactor operating conditions.

Annealing of the Belgian Reactor BR-1 is being considered, and some data on the accumulation of stored energy in graphite as a function of irradiation temperature and integrated neutron flux were presented in the form of curves, $E = f(\phi, T)$, at a Harwell conference[36] on Mar. 17 and 18, 1958. From these curves, an empirical expression was derived which gives the stored energy as a function of fractional powers of the integrated neutron flux and irradiation temperature. With this empirical expression and the best-known temperature and irradiation flux history of BR-1, estimates were made of the stored-energy growth in the BR-1 reactor during various periods in 1959. Since the power level and the temperature history of the reactor were altered during the first two years of operation, it was necessary to make the calculations in several steps.

The results of the calculations showed that the BR-1, if it were annealed under the same stored-energy conditions as the BNL Graphite Reactor, would be annealed every one and one-half years, as compared with every four months in the BNL reactor. The difference between these two periods is explained by the fact that, in the BNL reactor, the highest flux region is also the lowest temperature region; therefore an unfavorable condition exists in the BNL reactor.

It was also pointed out that, according to the calculation, the maximum stored energy accumulated during the first two years of operation of the BR-1 would exceed 30 cal/g and that this value would increase during 1959 and 1960, when the irradiated dose rate would be considerably higher. It was also noted that little experimental information is available, and therefore the results of this type of calculation are not precise.

Actual calorimetric measurements of the BR-1 graphite will provide the real evidence for judging this kind of empirical calculation. In any event, it would be unfair to criticize this analytical effort since it was founded on the tenuous experimental data available at the time. It is clear that the temperature configuration of the BR-1 pile was far from comprehensive; therefore, although the temperature was actually well considered in the analysis, the stored-energy evaluation of the reactor could not be expected to be any better than the temperature data. Furthermore, it has not yet been established that the graphite-stored-energy build-up is not dependent on the fast-neutron-flux spectrum and dose rate; for this reason, the correlation of data from different reactors may not be easily accomplished. (M. C. Wittels)

Reactor Kinetics Calculations

The mathematical model employed in reactor kinetics studies depends on the information de-

sired. Generally, if over-all reactor safety is being studied, the conventional space-independent kinetic equations are used on the basis that the gross reactor behavior given by these equations is adequate in view of the uncertainty associated with the value of the reactivity addition; because of this, the maximum plausible reactivity addition should tend to be on the high side. When more detailed information on reactor behavior is required, it may be necessary to consider more complicated mathematical models which give better estimates of reactivity additions and which take into account the spatial variation of neutron density with time.

This Review considers some calculations concerning reactor safety, the interpretation of reactor behavior, the adequacy of certain mathematical models in kinetics studies, the reactivity associated with physical changes, and the influence of spatial variations on reactor behavior. In the first five papers,[37–41] which discuss various phases of reactor behavior, the conventional space-independent kinetic equations are considered. The sixth paper[42] deals with the use of the perturbation theory and the concept of assuming the reactor to be on a stable period to calculate reactivity changes associated with physical changes. The seventh paper[43] discusses the time-dependent diffusion equation and determines the time required for the flux spatial distribution to reach its asymptotic shape following a change in neutron-poison concentration. The eighth paper[44] considers different regions of a reactor and combines flux distributions (obtained from criticality calculations) with conventional kinetics equations to study reactor behavior.

Space-independent Kinetic Equations

Hurwitz[37] considers the approximate analysis of reactor start-up incidents. Three phases are associated with an incident: in the first phase, the reactivity increases to the point where the reactor is critical; in the second phase, the reactivity continues to rise above delayed critical at a uniform rate, and the power increases until a level is reached which trips the safety circuit; the third phase comprises the time interval between tripping of the safety circuit and the decrease of the reactivity by the insertion of control elements. Hurwitz discusses phases 1 and 2 and considers the period of the reactor at the safety-trip power level as a function of this power level when reactivity is added linearly with time. The conventional space-independent kinetic equations are employed, and the reactor is assumed to have no inherent control characteristics (control would be by external means at the time the trip power level is attained). Previous studies have considered numerical solutions to this problem and elaborate analytical solutions. However, the approach taken here is the use of approximate methods to obtain simple analytical relations.

As the reactor becomes critical during phase 1, the power level is that associated with the source strength times the prompt multiplication times a factor associated with the build-up of delayed-neutron emitters. This latter factor is not a sensitive function of the rate of reactivity increase, and it does not contribute an important part to the total power increase during phases 1 and 2; it was obtained on the basis of a single group of delayed neutrons and a zero lifetime for prompt neutrons as a function of the rate of reactivity increase and the effective decay constant associated with the delayed-neutron precursors.

During phase 2 the power level at time t, $N(t)$, is given by

$$N(t) = N_0 f(t) \exp \int_{t_0}^{t} p(t')\, dt' \qquad (3)$$

where N_0 is the power level at delayed critical, p is the inverse reactor period, and t_0 is the time at which the reactor becomes delayed critical. The factor $f(t)$ is determined for low rates of reactivity addition by considering the delayed-neutron precursors to be in quasi-equilibrium with the prompt neutrons. For relatively high rates of reactivity addition, $f(t)$ is determined by treating the delayed-neutron source as a constant.

At the time N reaches a value which trips the safety circuit (end of phase 2), Eq. 3 provides a convenient means of estimating the associated reactor reciprocal period, p_T. Approximate formulas are given[37] for p_T when the final reactivity is near prompt critical and also when the final reactivity is far above prompt critical. In obtaining these relations, p is required as a function of the reactivity addition; for reactivity additions near prompt critical, p is given by an exponential function of the reactivity addition; for reactivity additions substantially above prompt critical, p was taken to be proportional

to the prompt reactivity divided by the prompt-neutron lifetime.

For the stated conditions, the reviewer feels that these relations should help provide an over-all picture of certain areas of reactor safety analysis. The understanding associated with this over-all picture will facilitate the choice of a specific reactor; but, once the specific reactor has been chosen, it should be analyzed by more detailed methods, particularly if large amounts of reactivity are involved. It would be desirable to limit the minimum period to values such that the time required to return the reactor to a subcritical condition is short compared with the minimum period. When the reactor is very much above prompt critical, however, such fast response time cannot be reasonably achieved. Under such conditions, considerable effort should be associated with determining the energy released during the time interval following tripping of the safety circuits.

Consideration is also given by Hurwitz[37] to the total energy release associated with an instantaneous reactivity addition when the reactivity reduction is a linear function of the total energy generated. Analytical results were obtained with the use of the conventional reactor kinetics equations. The total energy release was the same for a given instantaneous reactivity addition whether or not delayed neutrons, as such, are considered. The time variation of the power, however, is a function of the delayed-neutron parameter values. A linear rate of reactivity addition was also considered; for this case, the net reactivity oscillates with time. If reactivity is added at a linear rate until the reactivity addition is some value above prompt critical and the subsequent addition rate is zero, the total energy release is determined by the total reactivity addition and the power coefficient of reactivity.

Wilkins[38] derives simple analytical expressions that estimate the values of the prompt-neutron density, the delayed-neutron-emitter density, and the reactor period at the instant it becomes prompt critical; the reactor is initially subprompt critical, and the reactivity increases linearly with time. The conventional space-independent kinetic equations are used. Thus the problem considered here is essentially the same as that discussed by Hurwitz,[37] if the trip power level is taken to be that at prompt critical. The results obtained are useful in understanding how certain parameters influence reactor behavior at prompt critical. For a linear rate of reactivity addition, the results given here facilitate evaluation of the initial conditions associated with reactor behavior in the region above prompt critical. No inherent reactivity compensation was considered in the analysis. Thus, in applying the results, the power level associated with prompt criticality should be low enough so that reactivity changes associated with an inherent power coefficient of reactivity are small. This would apply in start-up incidents.

The behavior of the reactor power following activation of control mechanisms determines, to a large extent, the safety of the reactor. Related to this is the associated rate of reactivity reduction required to keep the reactor power from exceeding a safe value. This latter aspect has been considered by MacPhee[39] for cases in which the reactivity addition is appreciably more or appreciably less than one dollar. The conventional space-independent kinetic equations are used; reactivity is assumed to be added instantaneously at zero time, and the response of the reactor is obtained as a function of the rate at which reactivity is removed from the system. This situation relates to a reactor which has a zero-power coefficient of reactivity and is controlled by external means. The reactivity time behavior associated with keeping the maximum neutron level below some maximum value is discussed for the two types of cases. Most of the investigation is analytical and applies to one effective group of delayed neutrons (the decay constant associated with the group of delayed-neutron precursors is not the conventional one). It was found that, when the amount of reactivity added is less than one dollar, the peak reactor power can be limited to some maximum value by relatively small rates of reactivity removal. These required rates would be a few tenths of 1 per cent $\Delta k_e/\text{sec}$. If the amount of the reactivity added is above prompt critical, the rate of reactivity decrease has a marked influence on the peak reactor power. For large reactivity additions, the peak power is that associated with a reduction in reactivity such that one dollar is present in the reactor, so long as the delayed neutrons do not contribute appreciably to the power at the time of peak power; this is apparent from the kinetic equations.

Machine calculations utilizing five and six delayed-neutron groups gave good agreement with the effective one-group results. However,

one delayed-neutron group cannot adequately represent five or six groups for all conditions; therefore, this agreement appears to be fortuitous. Nevertheless, the general results should still be true since the effective decay constant for the cases considered would not change enough to significantly affect the required rates of reactivity removal.

The energy release is normally a more significant parameter than the peak power. However, if the heat-removal system were conservatively designed, the operating power level would be significantly lower than the safe power level at which the reactor could operate; for high-performance reactors, such a condition could not be tolerated since the reactor would operate at the highest permissible heat flux.

Although the space-independent kinetic equations have normally been used to predict reactor behavior following a reactivity addition, these equations are also being used to calculate the effective reactivity associated with an experimental power trace. Such an application has been considered by Corben,[40] in which the net reactivity is determined in terms of the rate of change of reactor power and an integral over the past history of the power. This relation is then used to obtain the reactivity for specified variations in power with time. To analytically integrate the integral term, it is necessary, from a practical viewpoint, to consider the power-time relation to be a relatively simple function. Such studies improve the understanding of the reactivity changes associated with power variations and provide a means of checking the results obtained from a machine numerical calculation. In general, however, it appears that numerical integration by machine of the integral term is required to consider the power-time relations normally associated with power excursions.

In treating the space-independent kinetic equations by analytical means, it is convenient if fewer than six delayed-neutron groups can be used since the equations are then easier to manage. Because of the associated simplicity, many analytical studies consider only one effective delayed-neutron group to facilitate management of the equations. To investigate the validity of reduced neutron-group representation, Skinner and Cohen[41] compared results obtained from one-, two-, and three-group representations with those based on six delayed-neutron groups. The fraction yield and decay constant associated with the effective neutron groups were determined from the six-group constants on the bases that (1) the amplitude of the transfer function at small angular frequencies and low power be satisfied, (2) the low-frequency low-power behavior of the phase shift be adequately represented, and (3) the behavior in the transition region (between the frequency range governed by delayed neutrons and that governed by the prompt-neutron lifetime) be properly represented. The validity of the various group representations was studied by investigating the in-hour relation, the behavior of the transfer function, and also the transient system response to both positive and negative step inputs of reactivity. The results obtained indicate that the two- and three-group models offer a significant improvement over the one-group model. However, the three-group model would not normally offer enough improvement over the two-group model to justify its use in many cases. If large numbers of kinetic calculations were to be considered, two-group representations (rather than six-group representations) would save machine time. However, in general, the reviewer feels that it is doubtful whether the number of cases involved in an actual parameter study would be so large as to make the savings in machine time worth the uncertainty associated with the results. In analytical studies, the increased accuracy associated with two-group representation over that associated with one-group representation appears worth the additional effort.

Reactivity Changes Associated with Physical Changes

Reactor safety studies require knowledge of the maximum amount of reactivity which can be added to the reactor; this maximum value is often significantly above prompt critical, and perturbation theory may not adequately predict the reactivity addition associated with some physical change. A related question is the reactivity worth of control rods; in this case, the change in reactivity associated with rod movement can be appreciably more than one dollar.

To treat the case where perturbation theory may not adequately predict reactivity changes, the concept of dynamic reactivity has been introduced. Simmons[42] considers a dynamic reactivity, which is defined (relative to prompt

critical) as $\Delta k = -l\alpha$, where α is the asymptotic (prompt) flux decay rate observed in a pulsed-neutron experiment and l is the mean lifetime for prompt neutrons in the reactor made critical by uniform subtraction of $1/v$ poison. This dynamic reactivity is based on the neutron production rate at a given time relative to the neutron destruction rate at the same time, whereas the conventional reactivity definition considers the neutrons produced in one generation relative to those produced in the previous generation. These two definitions coalesce when the reactor is critical. The conventional reactivity (assuming no delayed neutrons and considering a reactor that produces ν neutrons per neutron causing fission) is given by $(\nu - \nu_c)/\nu$, where ν_c is that value of ν required for criticality. From perturbation theory, $(\nu - \nu_c)/\nu$ also represents the product $\alpha' l'$, where α' is the persisting inverse period of the reactor and l' is the mean lifetime of prompt neutrons calculated on the basis that the reactor is critical. By analogy, the dynamic reactivity is given by αl, where the value for l is a function of the flux distribution associated with a reactor having an inverse period α. The value for l can also be interpreted as the reactivity coefficient for $1/v$ poison. This follows from an interpretation of the neutron balance equation; the same form of equation is obtained if that term associated with the time dependence (assuming the reactor is on a persistent period) is replaced by one involving a $1/v$ neutron poison (assuming a critical reactor).

Simmons[42] points out that a fundamental difference exists between the dynamic and static reactivity when the reactor is not critical and that the dynamic reactivity concept is more correct when large reactivity changes are involved. The calculational procedures associated with the dynamic reactivity essentially consider α as an eigenvalue (ν_c is the conventional eigenvalue), with l being a function of the flux distribution resulting from the α calculation.

The author also pointed out that, so long as the spatial flux distribution remains the same as that for the critical reactor, l represents the average time required for a prompt neutron to cause fission. It appears reasonable that this is also the meaning for l when the reactor is not critical. Under such circumstances, l for a given reactor should not vary significantly with changes such as an alteration in control-rod position; once l is obtained, α calculations (or data from pulsed-neutron experiments) can be utilized to determine the associated dynamic reactivity change.

Spatial Variation of Flux with Time

Although most kinetic studies assume the space-independent model, use of this model is not always justified. Recently, more emphasis is being given to the spatial variation of the flux with time. Two questions which arise are (1) how quickly does the spatial distribution essentially reach the equilibrium distribution and (2) how much do the equilibrium distributions vary with time during a transient in which spatial nuclear parameter values vary with time. Answers to these questions can only be obtained by studying specific cases.

The first question was investigated by Garabedian and Leffert,[43] who considered the diffusion equation with the source term determined by an arbitrary slowing-down kernel. The slowing-down properties were assumed to be independent of position. Under these conditions, the harmonics method was utilized, with the flux considered to be a function of position and time. In the harmonics method a determinant of infinite order must be considered. In practical applications this infinite-order determinant is approximated by one of finite order. The number of terms required for good accuracy cannot be rigorously specified; however, it is normally assumed that, if the answer is essentially the same for determinants of order 5 and greater, the results are adequate. This extended method of harmonics provides a technique for obtaining the transient flux shapes at any time during a rise in the power level, but it does not take into account the reactivity changes associated with the power rise.

The specific reactor studied was a three-region slab reactor containing (1) poison in the central region, (2) fuel surrounding the central region, and (3) reflector surrounding the fuel region. Graphite moderator was considered to be present in all three regions. The time behavior of the flux was studied for cases in which various amounts of poison were removed from the central region. No reactivity coefficients were considered, and one group of delayed neutrons was assumed. The results are therefore chiefly of use in determining the times required for the asymptotic flux distribution to be attained. Most of the reactivity changes considered were very large (greater than 10 per cent);

however, one case involved a 1 per cent change and one a 0.4 per cent change. For the cases considered, the longest time required to reach the asymptotic flux shape was 0.005 sec, and this was associated with a reactivity addition of about 50 per cent.

The asymptotic flux distribution calculated by the harmonics method was compared with that obtained by a two-group multiregion calculation (WANDA program), in which the asymptotic flux distribution was based on a critical-reactor calculation. The agreement between the two distributions was quite good (the average deviation in flux values was 0.4 per cent, and the maximum deviation was 2.6 per cent). In further harmonics calculations in which ν was decreased to yield a pseudo-k_{eff} value of unity, the resultant flux distribution gave even better agreement with the WANDA result (the average deviation was 0.29 per cent, and the maximum deviation was 0.56 per cent). Thus, much of the difference between the results of the first set of calculations was due to the difference in solutions obtained for time-dependent and steady-state models. However, the difference was small, which indicates that the asymptotic flux distribution based on critical-reactor calculations adequately represents the equilibrium distribution that would occur in a supercritical reactor. The stabilization time for the cases considered varied from about 0.002 to 0.005 sec. The smaller times were associated with the smaller reactivity additions. These times were in good agreement (±50 per cent) with those given by the product of the prompt-neutron lifetime and the number of iterations required in a machine solution of the critical-reactor problem, as proposed by Henry.[44]

It appears that the extended harmonics method applies primarily to the determination of stabilization times, and, for this application, it is a relatively convenient and accurate method.

The question concerning the variation of the equilibrium flux distribution with time was investigated by Curlee,[45] who outlined a procedure for determining the changes in flux distribution during a reactor transient, taking into account the effect of energy generation on nuclear properties and the effect of nuclear properties on energy generation. The reactor was assumed to consist of several regions, and the spatial flux distribution was determined on the basis of steady-state two-group diffusion calculations. This latter calculation implies that the flux shape adjusts to the steady-state value very rapidly; normally, this is true within a few prompt-neutron lifetimes. Thus, if the time interval of interest is long compared with the prompt-neutron lifetime, the procedure should be adequate.

Two specific problems were considered. In one, a large, bare slab reactor was considered. The reactor was broken into four regions in which average nuclear properties were assumed. The inlet temperature to each region was considered to be a variable with time. In the conventional approach the initial flux distribution would be assumed to be independent of time, whereas the magnitude of the flux would be a function of the average reactivity added to the system. In the study the temperature change in each region of the reactor was considered. Such an event would change the spatial distribution of the flux. By taking this into consideration, critical calculations were performed using a two-group multiregion diffusion-theory calculation (WANDA program). For different combinations of average temperatures in the four regions of the reactor, WANDA calculations gave values for k_e and average flux in the regions. These results were utilized in simpler form in the reactor-kinetic calculations by relating the reactivity to the temperatures in each region in a linear fashion and also by relating the average fluxes in each region in a linear fashion to the temperatures in the four regions. The amplitude function of the flux was then determined by the conventional kinetic equations, in which the reactivity and flux in a particular region were related to the temperature of that region. The average temperature for each region was determined by a separate set of heat-transfer equations in combination with the average flux in the region. The results for this case showed a significant change in the spatial flux distribution with a change in the coolant-inlet temperature.

In the second problem, a two-dimensional model of the Pressurized Water Reactor (PWR) was analyzed. The reactor consisted of an annulus of highly enriched seed, with natural uranium in both the inner and outer regions of the reactor. The variation of flux in the axial direction was not considered, but the variation around the annulus was taken into account. The transient behavior of the reactor was obtained following a drop in the inlet temperature associated with one corner of the fuel seed (one-fourth the total seed area). This behavior was compared

with that which occurred when the inlet temperature to all parts of the seed was dropped.

The steady-state criticality calculations for this PWR model consisted of seven solutions for different sets of average temperatures in each of the three regions. The solutions were obtained using two-group two-dimensional calculations based on diffusion theory (PDQ program). The results from these calculations were utilized in fitting the reactivity and the average flux in a particular region to the average temperature of the region. This was repeated to utilize simpler functional relations in the kinetics equations. These latter equations were employed in the same manner as those for the first problem. Lowering the corner coolant-inlet temperature 40°F increased the maximum flux level about 50 per cent more than lowering the coolant-inlet temperature 10°F over the whole seed region. It was found that the coolant-inlet temperature of the whole seed region would have to drop by about 22°F to cause a flux rise of the same magnitude as that obtained by dropping the inlet temperature 40°F in one-fourth of the seed.

The preceding calculational procedure depends primarily on the ability to compute detailed stationary solutions, with the kinetic calculations being relatively simple. Because of the assumption that the stationary state exists at all times with regard to spatial distribution, this approach may not accurately predict transients associated with very short periods. However, under most operating conditions, changes of flux with time are relatively slow, and it appears that a detailed knowledge of the spatial flux distribution is an important consideration to be used in the design of the reactor. The approach used in this study appears to be well suited to investigating the operational characteristics of reactors. (P. R. Kasten)

References

1. *Nuclear Safety*, 1(1): 32-33 (September 1959).
2. J. D. Eichenberg et al., Effects of Irradiation on Bulk UO_2, WAPD-183, October 1957. (Unclassified AEC report.)
3. A. H. Booth, A Method of Calculating Fission Gas Diffusion from UO_2 Fuel and Its Application to the X-2-f Loop Test, CRDC-721 (AECL-496), September 1957. (Unclassified Canadian report.)
4. O. J. C. Runnalls, Uranium Dioxide Fuel Elements, CRL-55 (AECL-754), January 1959. (Unclassified Canadian Report.)
5.* J. A. L. Robertson et al., Behavior of Uranium Oxide as a Reactor Fuel, A/CONF.15/P/193.
6. W. B. Cottrell et al., Release of Fission Products from UO_2 (Trip Report to Westinghouse Bettis Plant, April 27-28, and Chalk River Laboratory, April 30-May 1). (Unpublished.)
7. A. B. Auskern, Westinghouse Electric Corporation, Bettis Atomic Energy Division. (Unpublished.)
8. A. H. Booth and G. T. Rymer, Determination of the Diffusion Constant of Fission Xenon in UO_2 Crystals and Sintered Compacts, CRDC-720 (AECL-692), August 1958. (Unclassified Canadian report.)
9. B. Lustman, Release of Fission Gases from UO_2, WAPD-173, March 1957. (Unclassified AEC report.)
10. R. W. Dayton and C. R. Tipton, Jr., Progress Relating to Civilian Applications During May 1959, BMI-1346, pp. 40-42, June 1, 1959. (Confidential AEC report.)
11. J. Schnizlein et al., in Chemical Engineering Division Summary Report for January-March 1958, ANL-5858, June 1958. (Confidential AEC report.)
12. J. Schnizlein et al., in Chemical Engineering Division Summary Report for July-September 1958, ANL-5924, December 1958. (Unclassified AEC report.)
13. J. Schnizlein et al., in Chemical Engineering Division Summary Report for October-December 1958, ANL-5959, March 1959. (Unclassified AEC report.)
14. R. K. Hilliard, General Electric Company, Hanford Atomic Products Operations, Richland, Washington, private communication to Argonne National Laboratory; see ANL-5858, p. 66. (Confidential AEC report.)
15. F. E. Littman and F. M. Church, Reactions of Metals with Oxygen and Steam, AECU-4092, Feb. 15, 1959. (Unclassified AEC report.)
16.* G. R. Hennig et al., Radiation Effects on the Oxidation Rate and on Other Chemical Properties of Graphite, A/CONF.15/P/1778.
17. L. Baker et al., in Chemical Engineering Division Summary Report for October-December 1958, ANL-5959, March 1959. (Unclassified AEC report.)
18. B. E. Hopkinson, Kinetics of the Uranium-Steam Reaction, *Journal of the Electrochemical Society*, 106(2): 102 (1959).
19. E. Hillner and J. N. Chirigos, Metal-Water Reactions as Applied to A1W Loss-of-Coolant Accident, WAPD-TM-144, December 1958. (Confidential AEC report.)
20. L. Corrsin, H. Steinmetz, and B. Marano, Sodium-Water Reaction Rate Studies, NDA-84-19, May 15, 1959. (Unclassified AEC report.)

21. H. L. Floyd, Combustion Hazards of Organic Coolants, HW-58413, Dec. 19, 1958. (Unclassified AEC report.)
22. R. H. J. Gercke and C. A. Trilling, A Survey of the Decomposition Rates of Organic Reactor Coolants, NAA-SR-3835, June 10, 1959. (Unclassified AEC report.)
23. *The Forum Memo to Members*, Atomic Industrial Forum, Inc., Vol. 6, No. 3, p.41, March 1959.
24. *The Forum Memo to Members*, Atomic Industrial Forum, Inc., Vol. 6, No. 5, p. 48, May 1959.
25. W. B. Cottrell, Ozone Explosion in Irradiated Cryogenic Apparatus and Implications Upon N. S. *Savannah* Gaseous Waste Disposal System, ORNL CF-58-7-97, July 24, 1958. (Unclassified AEC report.)
26. J. H. MacMillan, Babcock and Wilcox Company, Atomic Energy Division, 1958. (Unpublished.)
27. C. Dalmasso and G. F. Nardelli, The Wigner Release in Graphite-moderated Reactors, *Énergie nucléaire*, 6(5): 307; also, 6(6): 376 (1959).
28. R. W. Henson and J. H. W. Simmons, An Adiabatic Rise Calorimeter for Measuring Stored Energy in Irradiated Graphite, AERE-M/R-2564, February 1959. (Unclassified British report.)
29. J. H. W. Simmons, Great Britain Atomic Energy Research Establishment, Harwell, 1959. (Unpublished.)
30. NPPC Wigner Release Method, *The Forum Memo to Members*, Atomic Industrial Forum, Inc., Vol. 6, No. 5, p. 49, May 1959.
31. T. Bishop, The Windscale Atomic Piles, *Metal Progress*, 76(1): 105 (July 1959).
33.* A. H. Cottrell et al., Theory of Annealing Kinetics Applied to the Release of Stored Energy from Irradiated Graphite in Air-cooled Reactors, A/CONF.15/P/2485.
34. L. E. Stanford, Oak Ridge National Laboratory, 1959. (Unpublished.)
35.* R. W. Powell, R. A. Meyers, and R. G. Bourdeau, Control of Radiation Effects in a Graphite Reactor Structure, A/CONF.15/P/462.
36. G. Penelle, BR-1: Calculation of the Energy Accumulated in the Graphite by the Wigner Effect, Centre d'Étude de l'Énergie Nucléaire, Brussels, Sept. 4, 1958.
37. H. Hurwitz, Jr., Approximate Analysis of Reactor Start-up Incidents, *Nuclear Science and Engineering*, 6: 11-17 (1959).
38. J. Ernest Wilkins, Jr., The Behavior of a Reactor at Prompt Critical When the Reactivity Is a Linear Function of Time, *Nuclear Science and Engineering*, 5: 207-214 (1959).
39. J. MacPhee, On the Termination of Reactor Excursions, *Nuclear Science and Engineering*, 5: 273-284 (1959).
40. H. C. Corben, The Computation of Excess Reactivity from Power Traces, *Nuclear Science and Engineering*, 5: 127-131 (1959).
41. R. E. Skinner and E. R. Cohen, Reduced Delayed Neutron Group Representations, *Nuclear Science and Engineering*, 5: 291-298 (1959).
42. B. E. Simmons, The Dynamic Reactivity Interpretation of Pulsed Neutron Measurements, *Nuclear Science and Engineering*, 5: 254-256 (1959).
43. H. L. Garabedian and C. B. Leffert, A Time-dependent Analysis of Spatial Flux Distributions, *Nuclear Science and Engineering*, 6: 26-32 (1959).
44. A. F. Henry, The Application of Reactor Kinetics to the Analysis of Experiments, *Nuclear Science and Engineering*, 3: 61 (1958).
45. N. J. Curlee, Jr., Nonseparable Space-time Transients Resulting from Changes in Inlet Coolant Temperature, *Nuclear Science and Engineering*, 6: 1-10 (July 1959).

*Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, September 1958. (Individual papers are available from the Office of Technical Services, Department of Commerce, Washington 25, D. C. Bound volumes of papers are available from United Nations, Office of Public Information, New York.)

Aseismic Reactor Design

The review of building designs from a safety standpoint frequently introduces the consideration of forces on the structures which would result from ground motion in the immediate vicinity during an earthquake. An excellent introduction to the study of earthquakes and earthquake-resistant design has recently been published.[1] In the usual procedure, the designer familiarizes himself with the seismic activity of the area from records gathered by local seismologists and the U. S. Coast and Geodetic Survey and with the geological features of the site, such as structure, composition, and faulting. Since prediction of earthquake frequency (eventwise) or intensity is impossible, the largest precedent shock usually becomes the design criterion. Studies of the transmissibility of earthquake shock waves make it possible to evaluate the hazard associated with distant earthquakes.[2]

Earthquake intensities are estimated on either the Rossi-Forel scale (maximum X) or the more meaningful Modified Mercalli scale of Wood and Neumann (maximum 12). Intensities on either of these scales can be expressed as ground acceleration in fractions of g (the acceleration due to gravity), and it is this acceleration which gives rise to the forces mentioned above. The response of the structure under consideration to seismic waves depends on the natural frequency, or period, of the structure and on the spectrum intensities of the forcing function. Biot[3] analyzed typical earthquake records and determined the relation between the maximum accelerations recorded and the periods associated with them. The only remaining factor is the calculation or prediction of the natural period of the proposed structure.

A major contribution to the subject of aseismic design of structures is an article by Tsui[4] which develops lateral force design coefficients as a function of the rigidity of structures. Although the reviewer feels that the application of this article to reactor design goes beyond the original intent of the author, other authors have made reference to, or quoted from, the article, and its application has been proposed in connection with aseismic reactor design.

In Tsui's discussion,[4] base shear (defined as the lateral force transmitted to the structure from the ground) and the distribution of the lateral force are the parameters. The maximum base shear as a function of the undamped fundamental period of a one-mass structure is defined as the response spectrum. A one-mass system was idealized, and the El Centro Accelerogram of the Imperial Valley earthquake on May 18, 1940, was chosen for the forcing function. The equations of motion for this simple system were set up and solved on an analog computer. The acceleration response spectra were determined for degrees of damping amounting to 0, 5, 10, and 20 per cent of critical. The analysis was then restricted to types of structures in which the mass distribution was uniform, and these structures were categorized according to their deflected configuration. The deflections induced by moments and shears were considered equally divided, and an average base shear vs. fundamental period curve was given for damping factors of 5, 10, and 20 per cent of critical. Tsui selected 10 per cent damping as a standard, for which the maximum base shear is 0.67 times the total weight for the one-mass system and 0.47 for the equivalent one-mass system, with an associated fundamental period of 0.45 sec. These values were considered to be too high, and, on the basis of the velocity ratio of the El Centro earthquake to that of the average of 14 other earthquakes, the ordinates were reduced by 0.483. The resulting curve was designated as applying to the Zone 3 areas of the Equal Seismic Probability Map in the Uniform Building Code.[5] The values for Zone 3 were halved for Zone 2, and they were halved again for Zone 1.

Tsui proposed that the provisions outlined in his article be adopted as code criteria, and it should be understood from the outset that code provisions are *minimum* acceptable standards. The design of the nuclear power station[6] to be built in Japan by the British was based, in most cases, on forces three times as large as those of the Japanese building code. The increase in the expected cost of this reactor from earthquake considerations is estimated to be 1 to 3 per cent of the total cost.[7] Reactors generally are neither one-mass systems nor uniformly distributed systems. Instead, the system usually consists of several relatively massive components dynamically coupled. Biot[3] pointed out the hazard of either assumption when unjustified.

The large static or pressure loads usually involved require highly rigid structures, implying, but not assuring, that the more massive components will follow the ground motion closely. If this is the case, the relative displacement between reactor components and external connections will be low, and the stresses induced will be low compared with other considerations. A consequence, however, is that dynamic loads on the internal reactor components are essentially the same as the forcing function; i.e., no damping has occurred. Considering, in particular, graphite-moderated reactors, allowance must be made for thermal expansion and Wigner growth and contraction. Gaps between graphite blocks preclude consideration as a unit mass unless unit mass behavior can be assured at room temperature, at operating temperature, and after continued irradiation. Alignment of fuel- and control-rod channels through the moderator for the lifetime of the reactor must be incontrovertible.

As a result of frequent misinterpretation, the U. S. Coast and Geodetic Survey, source of the Equal Seismic Probability Map referred to earlier, withdrew the map about five years ago in favor of more detailed attention to individual areas on a current basis. Since that time, the map has been retained by the Pacific Coast Building Officials Conference. The U. S. Coast and Geodetic Survey[8] believes that conferences with designers intending to use this map are warranted to reduce the danger of misinterpretation. Alvy[9] recommends that the earthquake intensity to be applied to reactor installations anywhere be Mercalli 5, as an absolute minimum.[9] On the other hand, application of the Equal Seismic Probability Map, without consideration of nuclear hazards, to large areas of the United States would result in no earthquake provisions whatever.

Under certain conditions, external restraints added specifically from seismic considerations can become battering rams if a gap is necessary to accommodate thermal expansion. The possibility of resonance exists in the path from the earthquake epicenter to the site, in the coupling between the foundation and support structure, and in the coupling between different components of the reactor.

Solely on the basis of the additional cost of aseismic design, the reviewer believes that a dynamic analysis of the reactor as a complete system having the appropriate degrees of freedom, entirely within the elastic range, can be justified. Tests to provide experimental verification should include the effects of thermal expansion and, where appropriate, predicted Wigner growth and contraction. Such an approach would provide detailed knowledge of both the dynamic behavior of the system and the hazards involved and would be preferable to the application of any code criteria, whether or not a scaling factor is applied.

(Boyce Y. Cotton)

Auxiliary and Emergency Electric Power Systems

The operating characteristics of a reactor demand exacting reliability from auxiliary and control power sources. Total failure of these sources may jeopardize the reactor. To avoid unnecessary shutdowns, alternate power sources are provided for auxiliary equipment and the control system. Although the designs of these alternate power-source systems may vary somewhat from installation to installation, they must satisfy the same requirements to assure safe operation of the reactor at all times. They must provide uninterrupted power for the safety system and for vital equipment, such as instrumentation and controls, and must supply power after shutdown for the operation of after-heat-removal equipment.

Three independent primary power sources are required to give good assurance of continued operation and sufficient power after shutdown. Provision for three such independent

primary power sources for the auxiliary system seems to be the accepted approach in most reactor plants.[10–13] All three sources normally supply the load requirements through individual step-down transformers. Two of the sources constitute high-voltage transmission lines, and one is the plant generator. One of the power sources is normally used as the "stand-by" and is automatically connected to the load in case of failure of either or both of the other two sources. In the N.S. *Savannah*, where the power supply is self-contained, two turbine generators provide the primary source for the auxiliaries, and diesel generators are available as a stand-by source.[14]

The distribution of the auxiliary power to the equipment is effected in different ways. The Enrico Fermi Atomic Power Plant[11] has three primary power sources connected to a single solid bus arrangement, with two of the sources supplying power simultaneously and the third in stand-by status. The bus provides continuous power to vital equipment of intermediate voltage level, such as large coolant pumps. When either source is disconnected from the common bus, the unaffected source will assume all load requirements without interruption until the stand-by power source is automatically connected. This scheme is simple and attractive. The main objection is the possibility of a bus failure and the resulting loss of all auxiliary power. However, with careful design, such a possibility is extremely remote.

By splitting the single bus arrangement into two sections, as in the Dresden Nuclear Power Station, the severity of a bus failure is minimized.[10,15] Also, with this design it is possible to divide vital loads in such a manner as to permit operation of partial loads during the failure of either feeder until the automatic operation of the power-transfer breakers or of the bus tie-breaker is accomplished. Although some flexibility is gained with the two-section buses, there exists the possibility of failure to transfer when required or of a delayed transfer. It may then be necessary to drop certain nonessential loads to enable the vital loads to start under acceptable voltage conditions with the short recovery time necessary to avoid a possible shutdown. The N.S. *Savannah* utilizes a double bus system with the tie-breaker normally closed, and each bus section is connected to a turbine generator.[14] This provides the advantages of a single bus arrangement. Bus faults are localized by opening the tie-breaker and automatically transferring all vital loads from the affected section to the unaffected section.

A three-section bus arrangement with tie-breakers normally open and each section fed from one independent primary source introduces further flexibility in vital load division in the Yankee Electric Plant.[13] A bus fault is unlikely to call for a complete reactor shutdown. However, in the case of primary source failure, the complexity of switching is increased. Temporary loss of voltage to a vital partial load is unavoidable during a source failure; and, under certain operating conditions, this may require a shutdown. The possibility of failure to transfer also exists.

A four-section bus arrangement which is in operation in the Shippingport Atomic Power Station[16] makes use of only two primary sources. Each source feeds its own three-winding transformer, each secondary winding of each transformer being connected to one section of the bus. Normally open tie-breakers provide for interconnection of the bus sections. When either source fails, an automatic transfer to the unaffected bus is accomplished. Experience has indicated that a conventional open-circuit transfer will cause shutdown of the reactor under certain load conditions; therefore, the auxiliary transformers are temporarily paralleled before the faulty section is disconnected. An additional transmission line is provided as an emergency source, but its use is left to the discretion of the operator. Failure of any one winding in a transformer in this arrangement can cause a plant shutdown because the affected transformer must be isolated from the electrical system before the load is transferred to another source of power, which temporarily disconnects vital loads.

Complete loss of auxiliary power requires a reactor shutdown. Vital equipment is then transferred to an emergency supply, either a diesel generator[10–12] or an emergency transmission line.[10,16] Other systems rely entirely on the three primary power sources and power-distribution arrangement to meet the needs.[13] In the Yankee Atomic Electric Plant,[13] high-speed relaying and transfer switching are essential to meet all possible fault conditions. Failure of a diesel to start or a delay in its

operation could be serious. To partially overcome this possibility, Frick[12] has proposed to make use of the large coastdown inertia of the main turbine generator. This is accomplished by providing an auxiliary generator on the same shaft with the main unit. The auxiliary generator will then supply the emergency power needed during the coastdown until the diesel or some other emergency power source can take over the duty. This arrangement has definite possibilities. A somewhat different situation exists in the N.S. *Savannah*.[14] A third emergency diesel generator will provide the power during a complete failure of the primary source and auxiliary diesel generators.

Reactor safety systems normally require 120-volt a-c power sources. The Dresden Station[10] utilizes two d-c–a-c motor-generator sets connected to sectionalized station power buses on the source side and to safety buses on the load side. To avoid nuisance scrams caused by voltage dips, the units are equipped with flywheels. Two parallel safety systems are provided, with the scram circuit designed to respond to dual-source failure only. The Enrico Fermi Plant[11] utilizes two station batteries to drive dual sets of d-c–a-c motor generators which supply separate buses for the safety systems. The battery chargers are supplied from one of several station power sources, and the batteries float on their respective buses, to which the d-c–a-c units for the safety system are connected. In the opinion of the reviewer, this represents a more reliable system than the previous one. A variation of this method is the use of only one d-c–a-c motor-generator set with provision for manual transfer to an alternative a-c source when desired, as in the Yankee Plant.[13] In this case, the two sources are momentarily paralleled during the transfer. The vital instrument power for the Shippingport Station[16] depends on two sources from the station power, with an automatic transfer arrangement, and one d-c–a-c motor generator with automatic transfer equipment as an emergency supply in case of complete source failure. In the N.S. *Savannah*,[14] two battery-fed d-c–a-c motor-generator sets provide the power source for the critical instrumentation and controls. One of the sets is considered a stand-by. From the information given, however, it is not possible to determine whether a transfer of units will cause a temporary interruption of critical loads. (E. Vincens)

Xenon Instability

High-flux reactors maintain criticality only so long as they can continue to burn out the xenon poisoning grown in from the fission-product I^{135}. The problem of maintaining enough excess reactivity to override xenon build-up after a brief shutdown has been dealt with in reactor control systems almost from the beginning of nuclear technology.

With the advent of physically large power reactors that run at high fluxes, the xenon-poisoning problem has assumed a new dimension. In this context, "large" reactors may be defined as those reactors whose linear dimensions are very much greater than their neutron migration length, and therefore there is no intimate intermingling of neutrons from opposite ends of the core. In large reactors it is possible for the flux in an isolated region to rise or fall independently of the flux in the rest of the reactor. Such an occurrence could arise from movement of a local control rod, adjustment of an experimental facility, fuel loading, accidental plugging of a coolant channel, or from almost any event whose influence on the reactor is localized.

In a high-flux reactor, such an isolated rise or fall in the neutron population will increase or decrease the rate of local xenon burn-out and will tend to put the region affected on a positive or negative period. Conventional reactor controls respond to any such fluctuations by the insertion or withdrawal of control rods in the reactor as a whole so that the over-all multiplication of the reactor will be kept constant, although the originally perturbed region may (as a result of the control action) run at far above or below the design-point power level. Regardless of which sign the original disturbance takes, some region of the reactor will end up hotter than it started out. This may be disastrous in reactors that, for reasons of economy or performance, are running their fuel elements at mean temperatures close to melting, ignition, or other failure point. To add to the difficulties, high-performance reactors are typically loaded for a flattened flux, and such configurations are most susceptible to the formation of hot spots in the power distribution.

The problem of stability criteria has been studied in a number of quite similar papers[17–22] in which (1) nuclear criticality and xenon-

production equations are written for a model of greater or less complexity, (2) linearized perturbations are introduced, and (3) a condition for stability is developed. Of the references cited, only the KAPL report[22] uses a one-group analysis. For greater accuracy, of course, it is highly desirable to consider the effects of the fast flux. Neglecting all but the thermal neutrons is, however, a conservative calculation since the greater range of fast neutrons makes them an effective agent for the smoothing out of local peaks or valleys in the flux. Hence, a reactor for which a one-group calculation indicates stability will certainly prove stable in the multigroup case. The effects of temperature coefficients are coupled tightly to the effects of xenon build-up, and therefore the two must be considered together in any stability analysis. For simplicity, however, temperature effects are omitted in the following sample calculation abstracted from the WAPD report written by Wick.[19] A bi-infinite reactor is postulated, with spatial disturbances considered only in the y direction at time t. The criticality equation is

$$\frac{[\Sigma_a - \delta\Sigma_a(y,t) + \Sigma_X(y,t)]\ \phi(y,t) - D\nabla^2\ \phi(y,t)}{\epsilon p \nu \Sigma_f^{25}\ \phi(y,t)} = 1 \quad (1)$$

where Σ_a = the absorption cross section without xenon
$\delta\Sigma_a$ = perturbation in Σ_a, such as insertion of a control rod
Σ_f^{25} = fission cross section, assumed constant
Σ_X = absorption cross section due to xenon
ϵ = the fast-fission factor
p = resonance-escape probability
ν = average fission neutrons released per fission
ϕ = thermal flux
D = diffusion coefficient
∇^2 = Laplacian operator

The equations for I^{135} and Xe^{135} formation are

$$\dot{I}(y,t) = \gamma_I \Sigma_f^{25}\ \phi(y,t) - \lambda_I\ I(y,t) \quad (2)$$

and

$$\dot{X}(y,t) = \lambda_I\ I(y,t) - \lambda_X\ X(y,t) - \sigma_X\ X(y,t)\ \phi(y,t) \quad (3)$$

where I = concentration of I^{135}
X = concentration of Xe^{135}
γ_I = "direct fission yield" of I^{135}
λ_I = decay constant of I^{135}
λ_X = decay constant of Xe^{135}
σ_X = absorption cross section of Xe^{135}

Very small perturbations $\psi(y,t)$ and $\delta\Sigma_X(y,t)$ are considered, such that

$$\phi(y,t) = \phi(y,0) + \psi(y,t) \quad (4)$$

and

$$\Sigma_X(y,t) = \Sigma_X(y,0) + \delta\Sigma_X(y,t) \quad (5)$$

It is assumed that the core is circular in the y dimension and that its parameters repeat every $y = h$. Time and space separability are also assumed, and $\delta\Sigma_a(y,t)$ and $\psi(y,t)$ are expanded in series:

$$\delta\Sigma_a(y,t) = A(t) \sum_n a_n \sin \frac{2\pi n y}{h} \quad (6)$$

and

$$\psi(y,t) = \sum_n b_n(t) \sin \frac{2\pi n y}{h} \quad (7)$$

Combining Eqs. 1 to 5 in the form of the Laplacian transform yields

$$\sum_n \left\{ D\left(\frac{2\pi n}{h}\right)^2 + \frac{D\nabla^2\ \phi(0)}{\phi(0)} + \phi(0) F[s,\phi(0)]\ b_n(s) \sin \frac{2\pi n y}{h} \right\} = \sum_n \phi(0)\ A(s)\ a_n \sin \frac{2\pi n y}{h} \quad (8)$$

where

$$F[s,\phi(0)] = \frac{\delta\Sigma_X(y,s)}{\psi(y,s)}$$

$$= \frac{\lambda_I \sigma_X [\gamma_I \Sigma_f^{25} - \Sigma_X(0)] - \sigma_X \Sigma_X(0) s}{\sigma_X \lambda_I\ \phi(y,0) + \lambda_I \lambda_X + [\lambda_X + \lambda_I + \sigma_X\ \phi(y,0)]\ s + s^2} \quad (9)$$

The individual terms $b_n(s)$ are then

$$b_n(s) = \frac{\phi(0)\ a_n}{D\left(\frac{2\pi n}{h}\right)^2 + \frac{D\nabla^2\ \phi(0)}{\phi(0)} + \phi(0) F[s,\phi(0)]} \quad (10)$$

The equation

$$D\left(\frac{2\pi n}{h}\right)^2 + \frac{D\nabla^2\,\phi(0)}{\phi(0)} + \phi(0)\,F[s,\phi(0)] = 0 \qquad (11)$$

represents the characteristic equation of the nth mode of flux perturbation. Positive real parts of the roots of this equation indicate instability. Substituting the actual value of $F[s,\phi(0)]$ from Eq. 9 into Eq. 11 and putting the result into a quadratic equation in s gives

$$\left[D\left(\frac{2\pi n}{h}\right)^2 + \frac{D\nabla^2\,\phi(0)}{\phi(0)}\right] s^2 + \left\{\left[D\left(\frac{2\pi n}{h}\right)^2 + \frac{D\nabla^2\,\phi(0)}{\phi(0)}\right] \times [\lambda_X + \lambda_I + \sigma_X\,\phi(0)] - \phi(0)\,\sigma_X\,\Sigma_X(0)\right\} s + \left[D\left(\frac{2\pi n}{h}\right)^2 + \frac{D\nabla^2\,\phi(0)}{\phi(0)}\right] [\sigma_X \lambda_I\,\phi(0) - \lambda_I\lambda_X] + \lambda_I\lambda_X\,\Sigma_X(0) = 0 \qquad (12)$$

in which $D(2\pi n/\pi)^2$ is always positive and larger in magnitude than

$$\frac{D\nabla^2\,\phi(0)}{\phi(0)} = -DB_0^2$$

Hence, the coefficient of s^2 is always positive. A sufficient condition for instability is for the coefficient of s to be negative, or for

$$\frac{\phi(0)\,\sigma_X\,\Sigma_X(0)}{\lambda_X + \lambda_I + \sigma_X\,\phi(0)} > D\left(\frac{2\pi n}{h}\right)^2 - DB_0^2 \qquad (13)$$

where the initial flux is the fundamental mode, $B_0^2 = (2\pi/h)^2$, and the condition for stability vs. the nth mode becomes

$$\frac{\phi(0)\,\sigma_X\Sigma_X(0)}{\lambda_X + \lambda_I + \sigma_X\,\phi(0)} > D\left(\frac{2\pi}{h}\right)^2 (n^2 - 1) \qquad (14)$$

If the flux is very high, i.e., above 10^{15} neutrons/(cm^2)(sec), so that $\phi(0)\,\sigma_X \gg \lambda_X + \lambda_I$, the relations

$$D \cong M^2\Sigma_a$$

and

$$\left(\frac{\Sigma_X}{\Sigma_a}\right)_{\text{equil}} \cong 0.05$$

where M is the migration length, give the equilibrium condition

$$\left(\frac{h}{M}\right)^2 > 20(2\pi)^2\,(n^2 - 1) \qquad (15)$$

or

$$\frac{h}{M} > 28(n^2 - 1) \qquad (16)$$

This condition shows that above a critical size, measured in migration lengths, the reactor will oscillate in various spatial modes, the higher modes being less easily excited.

If the y dimension (circular) is broken into two regions and the diffusion equations are solved using the boundary condition that the flux is constant in one region, stability conditions result which show stability to be inversely proportional to the relative size of the flattened region. This property may reduce the stable size of such a reactor by a factor of 3, according to Randall and St. John.[18]

The stability equations developed in the references cited are, of course, chiefly useful for determining whether a potential hot-spot problem is present in any proposed reactor. Development of a control system to meet such a threat to reactor stability demands a multiregion machine computation of reactor kinetics as affected by xenon, temperature coefficients, and whatever local control mechanisms may be under consideration. Suitable computation methods have been devised for both digital[23] and analog[24,25] machines, and they should be used where generalized criteria indicate inherent xenon instability. (R. S. Stone)

References

1. R. W. Abbett, ed., Earthquake and Earthquake-resistant Design, in "American Civil Engineering Practice," Vol. III, pp. 34-02–34-23, John Wiley & Sons, Inc., New York, 1957.

2. F. Neumann, "Earthquake Intensity and Related Ground Motion," University of Washington Press, Seattle, 1954.
3. M. A. Biot, Analytical and Experimental Methods in Engineering Seismology, *Transactions of the American Society of Civil Engineers*, 108: 365 (1943).
4. E. Y. W. Tsui, Aseismic Design of Structures by Rigidity Criterion, *Proceedings of the American Society of Civil Engineers, Journal of the Structural Division*, Paper 1942, p. 81, February 1959.
5. "Uniform Building Code," 1952 edition, Volume I, Pacific Coast Building Officials Conference, Los Angeles. (See inside of back cover which contains a map of the United States showing zones of aseismic probability.)
6. K. J. Wootton, Designing an Earthquakeproof Nuclear Power Station, *The New Scientist*, 5(127): 913-917 (1959).
7. Japan Plans to Build Nuclear Reactor, *Modern Power and Engineering*, 53(4): 70-71 (1959).
8. Conversation with Leonard Murphy, U. S. Coast and Geodetic Survey, Washington, D. C., Aug. 26, 1959.
9. R. R. Alvy, *Civil Engineering*, Society News, p. 84, June 1959.
10. W. J. Shewski, A Look at the Electrical Features of Dresden Nuclear Power Station, *Power Apparatus and Systems*, 41: 62-67 (April 1959).
11. R. H. Logue, Electrical Engineering Aspects of the Enrico Fermi Atomic Power Plant, *Power Apparatus and Systems*, 41: 71-76 (April 1959).
12. R. E. Frick, Auxiliary Power Systems for Nuclear Plants, *Power Apparatus and Systems*, 41: 86-88 (April 1959).
13. E. T. Witt et al., Electrical Features of the Yankee Atomic Electric Plant, *Power Apparatus and Systems*, 41: 94-107 (April 1959).
14. J. H. MacMillan, ed., Babcock and Wilcox Company, Atomic Energy Division, 1958. (Unpublished.)
15. V. A. Elliot et al., The Dresden Nuclear Power Station, *Power Apparatus and Systems*, 40: 1367-1382 (February 1959).
16. H. G. Frus et al., Electrical and Control Features of the Shippingport Atomic Power Station, *Power Apparatus and Systems*, 40: 1487-1492 (February 1959).
17. A. G. Ward, The Problem of Flux Instability in Large Power Reactors, CRRP-657 (AECL-345), July 26, 1956. (Unclassified Canadian report.)
18. D. Randall and D. S. St. John, Xenon Spatial Oscillations, *Nucleonics*, 16(3): 82-86, 129 (March 1958).
19. R. S. Wick, Space- and Time-dependent Flux Oscillations (and Instability) in Thermal Reactors Due to Nonuniform Formation and Depletion of Xenon, WAPD-TM-138, August 1958. (Unclassified AEC report.)
20. J. G. Tyror, Great Britain Atomic Energy Research Establishment, Harwell, 1958. (Unpublished.)
21. J. G. Tyror, Controlling Instabilities in Graphite Power Reactors, *Nuclear Power*, 4(38): 94-97, 124 (June 1959).
22. J. C. Stewart, A Generalized Criterion for Xenon Instability, KAPL-M-JS-4, Dec. 18, 1958. (Unclassified AEC report.)
23. J. B. Callaghan et al., TURBO—A Two-dimensional Few-group Depletion Code for the IBM-704, WAPD-TM-95, November 1957. (Unclassified AEC report.)
24. A. C. D. B. Vianna, Control Aspects of Very High Flux Research Reactors (Thesis), ANL-6008, May 1959. (Unclassified AEC report.)
25. N. J. Curlee, Jr., Nonseparable Space-time Transients Resulting from Changes in Inlet Coolant Temperature, *Nuclear Science and Engineering*, 6: 1-10 (July 1959).

Decontamination Hazards

Decontamination of radioactive equipment is normally undertaken to reduce radiological hazards, and extensive efforts have been devoted to the development of equipment and processes for such work. The hazards present during the time the decontamination procedures are being carried out have, however, received little attention in the published literature. Therefore it is of interest to note that the radiation exposure records of ORNL show that, in most cases, workers receive more radiation during decontamination of equipment than during operation of the equipment under normal conditions. The subject of decontamination hazards should accordingly be given special consideration. No attempt is made here to review "normal" hazards that would be encountered outside the nuclear field. Thus, without intending to minimize their importance, hazards from chemical reactions (toxic effects, chemical burns, corrosion to the point of weakening equipment, and interaction with other chemicals), heat pressure, etc., are ignored, and only criticality and radiation hazards are considered.

In the current literature on decontamination, no mention is made of criticality hazards during decontamination operations. There is, however, reference to the necessity for decontamination of loops containing ruptured UO_2 fuel elements and plutonium fuel elements,[1] and therefore it is conceivable that a critical incident might occur during these operations. Although criticality hazards are very improbable during decontamination, the possible consequences of a criticality incident are serious enough to warrant the inclusion of such an incident as a separate item in any check list for establishing decontamination procedures.

Criticality control by geometry is probably the surest way to eliminate criticality hazards in decontamination operations. However, this would require infinitely safe dimensions for all vessels that could contain fissionable material, and such geometry would not always be feasible. One possibility, which was not suggested in any of the literature reviewed, is the inclusion of neutron poisons (such as boric acid) in decontaminating solutions; this might be worth considering for special applications. At present, it appears that complete reliance is being placed on operating control of criticality hazards during decontamination. Decontamination procedures should include safeguards that will lessen dependence on the judgment and ability of individual operators, and they must include adequate supervision of the work.

Of all the recent literature reviewed, the most comprehensive decontamination report is a United Kingdom Atomic Energy Authority document entitled "Recommended Reagents for Radiological Decontamination."[2] It is a combination catalog, directory, and instruction manual for British decontamination materials and methods as of 1958. Recommended reagents and procedures can be found for practically any situation from laundering of contaminated clothing to decontamination of stainless steel that has been heated to red heat. To keep the report concise, it was prepared primarily in outline form, and the only information included is that which is necessary to prepare and use the reagents. One serious omission is data relative to the effectiveness of each of the procedures. The closest the report comes to stating an expected decontamination factor is the statement (in "Method of Use" for one of the reagents): "The time of treatment will vary, but 15 min should give a very considerable reduction in contamination." Another serious omission is one which is common to most current decontamination reports — hazards. The report occasionally mentions toxicity and/or flammability; with these exceptions, the closest approach to a discussion of hazards is one statement in connection with strippable film layers that "cross contamination from one surface to another is almost entirely avoided."

Although none of the reports reviewed discussed decontamination hazards per se, many of them contained an occasional item bearing on the subject. For example, the report on the

decontamination program for the Army Package Power Reactor[1] (APPR) states that, for a successful decontamination program, "there must be suitable equipment to process large volumes of highly contaminated liquids," and recommends that future reactor designs include "provisions for isolation of contaminated components." It also recommends that future designs avoid stagnant areas where deposits may collect ("crud traps"). The primary purpose of this recommendation is to facilitate decontamination, but the elimination of such stagnant areas obviates the potential hazard involved in a possible increase in concentration (in the stagnant area) of either radioactive or fissionable material.

Another volume of the same report is entitled "Recommended Procedure for Decontamination of a Stainless-steel Steam Generator."[3] Estimates are given for the radiation levels of various portions of the equipment at the end of the life of the first APPR core, and it is stated that the decontamination solution activity may be as high as 20 $\mu c/cm^3$. It is obvious that adequate provision must be made for the protection of personnel both from high initial activity and from highly radioactive-waste solutions.

An interesting technique in the handling of radioactive solutions is given by Snavely and Manno.[4] Although the subject is "Handling Contamination in Spent-fuel Shield Water," the problem is analogous to the handling of water rinses in decontamination operations. In their process, the water is circulated through an ion-exchange resin prior to discharge. However, activity concentrated on the ion-exchange bed constituted a hazard to operating personnel. Consequently, a small ion exchanger (capacity, 0.6 ft^3) was placed ahead of, and in series with, the large ion exchanger. The small exchanger is shielded with 3 in. of lead. In 34 days of operation, 68,850 ft^3 of water was circulated through the two resin beds in series; during this period the activity of the large bed remained at 13 mr/hr, whereas the activity of the small bed increased from 0 to 260 mr/hr. The normal procedure is to discard the resin in the small bed when its activity builds up to about 300 mr/hr. With this procedure, the large exchanger requires regeneration only twice a year.

The comments thus far in this Review have been concerned with aqueous solutions. There is no intent to infer that there is any difference between aqueous and nonaqueous methods in the degree of hazard; the differences are merely in the details of the hazards. For example, the primary hazard associated with abrasive-blasting decontamination is the possibility of inhalation of radioactive dust. A report on "Equipment Decontamination by Abrasive Blasting in the Processing Refabrication Experiment"[5] indirectly discusses hazards and their alleviation. For dust collection, "air flowing at 1000 cfm is exhausted... through a bag house dust collector using Orlon bags with 352 square feet of net cloth filter area. The air is passed in succession through a glass prefilter and an absolute filter before it is exhausted to the atmosphere. The need for multiple filters in series was dictated by the necessity for meeting Health Physics standards and local code requirements." Most of the report is a comparative study of various abrasive materials and details of technique (such as angle of impingement). It was found that the dust-production rate is "dependent on the abrasive rather than other variables," and that the "dust-production rate with mineral abrasives is of the order of 10 to 100 times the rate with metallic abrasives." There was a gradual build-up of activity on the abrasive itself. Blasting with a contaminated abrasive against a clean target gave no evidence of transfer of activity to the target, but it did result in a "substantial decrease in the activity level of the abrasive.... Sampling of exhaust discharge air gave no indication of activity. Smears taken on the inside surfaces of the cabinet and ducting showed little deposition of activity." Studies showed that "the quantities of dust particles less than 1 μ in size are minute, so that commercial dry filtration equipment can be applied," and that air-borne contamination in the exhaust air was undetectable. For service where it is not feasible to handle contaminated abrasives, it has been suggested[1] that dry-ice pellets be used; the dry ice would evaporate and leave the contamination in a form which could be washed out.

The hazards involved in decontamination are, of course, determined to a large extent by the specific contaminant. This refers not only to the radioisotopes involved but also to their chemical and physical forms. For example, in a discussion of the removal of tritium contamination,[6] it is stated that tritium oxides (HTO, DTO, and T_2O) are up to 1000 times more hazardous than HT, DT, or T_2 because of differences in absorption rates. However, since tritium, a

beta emitter, is readily oxidized by atmospheric oxygen, "The protective measures taken for personnel working with tritium in any form should... emphasize the importance of controlling inhalation and absorption." Control and monitoring measures include isolation, adequate ventilation, protective clothing, and urinalysis. The report[6] states that there have been cases "in which a supposedly clean surface... has become recontaminated while standing in a tritium-free atmosphere. This phenomenon may be explained by the reverse diffusion, or 'bleed-out' of tritium activity from beneath the exterior of the metal back to the outer surface." This type of action constitutes a hazard associated with decontamination in that a "decontaminated" item may later be contaminated while being handled as if it were clean. Thus, for decontamination from materials involving only nonpenetration radiation, one should (1) be sure that the contamination is entirely on the surface, (2) be sure that there is no possibility of reverse diffusion, (3) decontaminate to a depth greater than that of any possible diffusion, (4) accelerate the reverse diffusion as a part of the decontamination procedure, or (5) maintain surveillance over the decontaminated item until it is certain that there will be no activity build-up.

To achieve *effective* decontamination, it is *desirable* that careful planning precede operations. To achieve *safe*, effective decontamination, it is *imperative* that careful planning, including consideration of all possible hazards, precede operations. (W. H. Carr)

Concrete Pressure Vessels

Vessels capable of retaining internal pressures of a few pounds to hundreds of pounds per square inch are needed by the nuclear industry for either the plant containment vessel or the reactor pressure vessel, or both. Since the costs of such vessels are a significant portion of the initial capital cost, there is great incentive to find more economical materials and methods of construction than those associated with the steel pressure shells currently in use. This incentive is particularly present with gas-cooled reactor power plants because of the inherently large size of these systems. Consideration of the use of concrete for such pressure vessels suggests that it may eventually find wider application.

An interesting and novel means of using prestressed concrete for the main reactor pressure vessel was demonstrated by the French in the G2 and G3 reactors at Marcoule.[7] Close scrutiny of this rather startling approach to pressure-vessel fabrication indicates that it possesses some definite merits.

The pressure vessels at Marcoule are horizontal cylinders 46 ft in inside diameter and 59 ft in length. The thickness of the walls is 10 ft. Each end is enclosed in a semispherical cupola, with the concave sides outward so that the action of the pressure keeps the concrete in compression. The cylinder is mounted on two bearer beams 197 ft long, 115 ft wide, and 29.5 ft high. The structure contains 27,500 yd^3 of concrete, 3000 tons of reinforcing steel, and 760 tons of prestressing steel strands.

The concrete in the cylindrical walls is maintained in compression by means of steel cables threaded through conduits precast into the concrete. The cables wrap around the shell through an arc of 250°, and they are anchored in a cavity extending along the bottom edge of the cylinder. The cables are posttensioned (i.e., stressed after the concrete is set and cured) so that each of the 797 strands is stressed to about 44.5 and 47.5 tons/$in.^2$ This is considerably less than the ultimate 90 tons/$in.^2$ guaranteed by the manufacturer and is considered rather low for prestressed concrete structures. On the basis of experience with large hydraulic dams, the tension force for each cable was chosen as 1200 tons.

The axial load of 45,000 tons on the end cupolas is held by means of longitudinal cables consisting of 715 steel strands (5 mm in diameter) bound together by steel strips into a unit. The radial component of this force was offset by means of steel hoops.

The effects of stress relaxation in the steel strands and concrete extrusion were countered by (1) overestimating the friction losses in the circumferential cables and (2) stressing to a higher value than required. The tension on each cable can be checked periodically and increased, if necessary, throughout the life of the reactor. If a cable is defective, it can be replaced at a later date; or, if an increase in pressure is desired, larger cables can be substituted. Corrosion of the cables has been guarded against by free use of lubricants in the cable and its

sheath and by heat generation due to gamma heating in the concrete, which keeps it warmer than the surrounding atmosphere.

Prestressing causes the concrete to be at its most highly stressed state when there is no internal pressure in the shell. This maximum stress is 1422 psi, which is lower than values used for many vault type dams.[7] Theoretically, as the internal pressure is increased, the compressive load on the concrete decreases and the tensile load on the steel increases until, at operating pressure, the complete load is on the steel and the concrete is unstressed. Because of the weak tensile strength of concrete, however, it is kept at some state of compression during reactor operation to ensure against effects of overpressure and unknown second-order effects not included in the design theory.

If the internal pressure is increased beyond the zero-load condition, the concrete will aid in containing it until a tensile stress of 569 psi is attained, at which time it will crack, and the whole load will be thrown on the prestressing steel. The steel wires should be able to hold this load until a very high stress is attained since they do not have a definite yield point.

Since concrete is rather permeable for this kind of service and since the effects of shrinkage and stress cracks are unknown, the inner surface of the vessel was lined with steel plate. This lining was shored up before pouring, and it acted as the inner form for the concrete. This plate provides an impermeable envelope for the cooling gas. In case an overpressure causes the concrete to be in tension, the concrete will crack and throw the complete load on the steel cables. The cracks will propagate, but, because of the large thickness of the concrete, it will break into large pieces that will still provide sufficient holding of the inner liner.

A disadvantage of the prestressed cylindrical structure is that all penetrations are crowded into the cupola ends. Individual penetrations for each of 1200 channels were cast into one end, and 14 39-in.-diameter tubes for supplying the reactor coolant, CO_2, were cast into the other end. The number of main coolant supply loops could be reduced by using a single cooling loop with a bypass for pressure-vessel cooling (this arrangement is planned for the Experimental Gas-cooled Reactor (EGCR) being built at Oak Ridge) instead of three separate ones (one each for two regions of the reactor, and one for shell cooling). It would be possible to install penetrations from the sides by anchoring the prestressing cables to a ring around the hole, but this might involve unwarranted increased complexity.

Concrete has also been suggested for use in conjunction with a containment scheme such as that used at the Oak Ridge Research Reactor.[8] In this case, the initial pressure surge following the primary system failure would be vented since it is known that a finite time is required for the fuel elements to overheat and rupture. Following this initial venting, the building openings would be sealed off, and a negative pressure would be maintained inside the structure by means of an exhaust fan blowing air through a filter and up a stack. Thus, all air leakage into the building would be inward, and all air would be filtered prior to expulsion to the atmosphere. Permeability of the building is not of too great concern.

In cases where complete containment is required, not enough is known about concrete at the present time to risk using it without an extra gas-sealing envelope, such as would be provided by a metal liner. Although some values have been reported[9,10] for the permeability of air in concrete, they range from 3.6×10^{-4} to 6.48×10^{-8} ft^3 of air per second through 1 ft^2 of concrete surface area 1 ft thick under a pressure of 1 lb/ft^2. These values show the extreme variability of concrete. Furthermore, use of any experimental value is questionable because of the effects of the hairline cracks that are always present in concrete structures.

Preliminary cost estimates for pressure containment shells, which must include the steel liner, show that they are only marginally cheaper than a steel shell. For instance, a steel-lined prestressed concrete pressure vessel for the EGCR (a vertical cylinder with a hemispherical head and flat bottom, 150 ft above grade, 50 ft below grade, and 115 ft in diameter, built to withstand 7 psi and 200°F with less than 2 per cent leakage per day) can be expected to cost only $15,000 less than a comparable steel vessel estimated[11] to cost $988,000. This steel shell cost does not, however, include the cost of the concrete shielding to be applied to the inside of the shell. The favorable price differential of a complete concrete shell would therefore be greater than this. It was proposed that this vessel be built in a somewhat different manner from the French vessel in that the prestressing would be applied by continuously

winding a wire around the circumference of a cylindrical shell and periodically fastening the wire to lugs cast into the concrete. After prestressing, the wire would be covered with about 1 in. of Gunite (sprayed mortar). This method seems somewhat less expensive than the more involved French method, but it does not have the advantage of cables that can be tested periodically for tension and replaced. Side penetrations could be installed simply by deflecting the wound wires around the penetrating member. This would impose second-order stresses on the concrete which would be difficult to estimate, but proposals have been made using this method.[12]

Some containment shell designs propose the use of a steel hemispherical dome tied to the concrete cylindrical wall. The containment shell for the Heavy Water Components Test Reactor (HWCTR) designed by Du Pont is to be of prestressed concrete below grade and steel above grade. It is a cylindrical vessel 70 ft in diameter and 125 ft in height. Sixty feet of the structure, which is made of 18-in.-thick concrete, is below grade and rests on a reinforced flat concrete slab.

The junction between the cylinder and the flat slab is sealed with a rubber ring embedded into both members. The concrete cylinder is poststressed by steel cables wrapped around it so that the concrete stress will be negligible when the structure is subjected to 30 psig internal pressure. There are 16 penetrations below grade, and they range up to 16 in. in diameter.

No liner is used in the concrete portion. The specified leakage rate is 1.0 per cent of the total volume per day. It is expected that the leakage rate through the concrete portion will be smaller than 0.10 per cent per day.[13] Leakage greater than this will not be of too great concern because there is a 3000-ft-radius exclusion area.

On the other hand, an experienced prestressed concrete designer and fabricator, The Preload Company of New York, feels that not enough is known about concrete to risk using it for a gastight application without a steel liner. The liner can, of course, be used in place of the inner forms, and the form cost can be deducted from the liner cost to obtain the net increase in price due to the liner. For instance, a ¼-in. steel liner can be expected to cost about $3.00/ft^2$, from which the cost of the inner forms, which might be considered to be worth about $1.25 to $1.50/ft^2$, can be deducted. Thus the net cost of the liner would be about $1.50 to $1.75/ft^2$ of wall.[14]

Various coatings have been developed to seal concrete surfaces. Although these were developed primarily for decontamination purposes, they would be applicable in some cases for leakage prevention. Polyethylene sheet can be laid and welded by using a hot iron at about 500°F. Polyethylene can also be sprayed on by blowing finely ground powder through a flame, but care must be taken to prevent the concrete from spalling (due to overheating) and causing a poor bond. Bubbles and pit holes can form in the plastic coating because of desorption and expansion of gases in the concrete.[15] Other similar coatings include vinyl paints, Teflon, paraffin, and epoxy and polyester resins with glass or asbestos mats. None of these materials is as reliable as sheet steel for gas containment, and further tests on these and other inexpensive methods of sealing concrete are needed.

The use of concrete is limited by the temperature that it can withstand. Tests of 5000-psi 28-day concrete briquettes exposed to temperature for 7 to 14 days show that, at 110°C, the strength increases to 7557 and 7231 psi, respectively. At 350°C the strength drops to 4156 and 3357 psi, respectively. At 500°C the strength is 2947 and 2935 psi. The increase in strength at 110°C can be attributed to acceleration of the aging process since concrete at room temperature increases in strength beyond the 28-day value throughout the first year.[16]

Concrete is a brittle material and possesses fairly low-energy absorption characteristics. It is 16 times less effective than steel in resistance to missiles for a given thickness.[17] Also, a prestressed structure has additional missile resistance because it has high-strength steel wire wrapped around the outside to keep the structure together. Concrete is weak in tension, and it must be steel reinforced, prestressed, or otherwise used in compression. Concrete manufacture is subject to deviations from specifications because of compromises forced by the weather; availability of specified constituents; and variations in compaction and proportions, unless closely supervised.

Furthermore, the design safety factors are lower in concrete than in steel pressure vessels. Allowable stresses in reinforcing rods are appreciably higher than those in mild steel vessels. A stress of 20,000 psi is allowed in reinforcing rods having a yield stress of 33,000 psi.[18] Stresses are even higher in prestressed structures, the 45 tons/in.^{2} used by the French being a conservative value, although it is one-half of the ultimate strength. Steel vessels, on the other hand, must be made of mild steel to be safe from effects of the brittle-fracture transition temperature. For type SA-201 grade B steel, which is commonly used for containment shells, the allowable stress at temperatures from −20 to 650°F is 13,750 psi; this steel has a yield stress of about 27,000 psi. For type SA-212 grade B steel used for pressure vessels, which has a yield stress of about 35,000 psi, the allowable stress is 17,500 psi.[19] In addition, steel shells require stress relieving between weld passes when plates of the sizes required for reactor pressure vessels are being welded. Furthermore, radiography of all welds is usually required if 100 per cent weld efficiency is expected. Prestressed concrete shells, however, are self-testing (i.e., the greatest stresses imposed on both the steel and the concrete occur during construction and decrease afterward due to stress relaxation and concrete plastic deformation). No special safety factors are used in normal prestressed-concrete tank design, but the ratio of the ultimate bursting strength to the design strength is estimated to be about 1.75. This could be increased to 2.5 to provide a structure equivalent to an all-steel shell designed under the ASME pressure-vessel code.[19] Deleterious effects of steel welding are avoided in reinforced- and prestressed-concrete structures because anchoring is usually done mechanically.

Although unknowns exist in concrete pressure-vessel technology for nuclear reactor applications, this could be a promising approach. Since concrete is trusted to hold water behind most of the large dams in the world, the rupture of some of which could cause disasters greater than a reactor incident, it should not be discounted categorically because of lack of familiarity among nuclear engineers with its application as a pressure vessel.

(M. H. Fontana)

Fuel-reprocessing Hazards

The various methods of fuel reprocessing* have certain hazards associated with them which must be provided for in the design of the facility. These hazards may be chemical, as well as nuclear. Where wet chemistry is involved, criticality situations are frequently of concern, whereas chemically initiated incidents generally lead to radioactive contamination and inhalation hazards. Recent reports[20–23] from Hanford, Brookhaven, and Oak Ridge have discussed nuclear and chemical hazards associated with various methods proposed for processing enriched-uranium fuels.

The Hanford study[20] is concerned with criticality problems in wet-chemistry processing of fuels having initial enrichments of ≤ 5 per cent U^{235}. Consideration is given to the critical parameters for heterogeneous fuel systems in water or in uranium solutions and homogeneous solutions of fuel and water, as well as to the use of boron poisoning to increase critically safe batch sizes. The types of fuel considered are uranium metal, uranium oxide, uranium-molybdenum alloys, and plutonium-aluminum alloys; the types of cladding considered for the fuel elements are aluminum, zirconium, and stainless steel.

A comparison of theory and experiment in the prediction of critical parameters is presented. This compilation of theoretical and experimental data makes this report a good reference guide for the design of nuclearly safe equipment. Good agreement between theory and experiment is indicated where data are compared on equivalent bases and is especially close at 3 per cent enrichment for 0.925-in.-diameter rods in water.

It is pointed out that the critical mass of a heterogeneous system is lower than that of a homogeneous system for enrichments less than 5 per cent. At about 5 per cent enrichment, the minimum critical masses are shown to be about equal for both systems, whereas above 5 per cent it is predicted that the homogeneous system will require the lower critical mass. Although the minimum critical mass for the homogeneous system is indicated to be about equal to that of

* Reviews of the methods are presented periodically in *Reactor Fuel Processing*, a quarterly technical progress review prepared for the U. S. Atomic Energy Commission by Argonne National Laboratory.

the heterogeneous system at 5 per cent enrichment, the *maximum* safe geometries for the heterogeneous system are still more restrictive beyond 5 per cent enrichment.

A comparison between uranium oxide in water and uranyl nitrate in water is presented to show the effect of nitrate poisoning on the critical concentration of uranium at various enrichments. The effect of the presence of nitrate is small since its poisoning effect is small; the effect becomes smaller as the enrichment of U^{235} increases. All critical concentrations are given as grams per liter of uranium, rather than as hydrogen-to-uranium ratios; therefore they do not apply if the solvent is other than light water.

Based on the theoretical study of critical concentration, the minimum enrichment for criticality in an infinite volume of a uranyl nitrate–water system is estimated to be about 1.92 per cent U^{235}. This compares with a preliminary experimental value of about 2.1 per cent U^{235}.

Two reasons given for not studying the effect of boron poisoning on systems containing 5 per cent enrichment are: (1) some of the presently contemplated processes for dissolving power-reactor fuels may result in volatilization of the boron, and (2) it is generally felt that boron in solution should never be used as a primary nuclear safety control. However, the large gains in plant capacity obtainable through the use of neutron poisons indicate that further work is justified in such critical-mass studies.

A Brookhaven report[21] contains short sections on both the nuclear and chemical hazards of a proposed plant for reprocessing the core fuel from the Enrico Fermi Fast Breeder Reactor. The fuel is 25 per cent enriched in U^{235}; therefore it was necessary to analyze carefully the criticality of the system during the process development and design.

Criticality control is obtained by geometry control in the dissolver and concentration control in the solvent-extraction equipment. The solvent-treatment system was made geometrically safe to eliminate accumulation of a critical mass in the event of a plant operational error. It was also recognized that it is possible to accumulate a critical mass in solvent-extraction equipment downstream from the dissolver if an operational error occurs. For example, if nitric acid were inadvertently omitted from the scrub stream, the system would be on total reflux, which could accumulate a critical mass in approximately 1.5 hr. It was considered doubtful that analytical results on end-stream samples would positively indicate that the solvent-extraction bank was on total reflux.

It was therefore suggested[21] that the only feasible method of detecting a potential nuclear reaction within a bank would be to use a neutron detector. It is recognized, however, that this would not prevent a critical situation from existing or warn the operator in time to allow him to prevent a nuclear incident since the rate of increase in uranium inventory may be as high as 450 g/hr. Further work would be justified to assure that absolute detection and control can be established. A technique to minimize the excessive concentration in the extraction equipment would be to employ in-line instrumentation to monitor continuously the nitric acid concentration.

The pyrophoric nature of zirconium-uranium alloys was also considered, and the existing experimental information on pilot-plant operation is given.[21] There has been no case of either detected pyrophoric surfaces or indications of rapid reactions during dissolution of the elements.

One of the ORNL reports[22] discusses a sequential evaluation of the nuclear and radiation hazards expected during the mechanical processing and storing of fuel elements from the Sodium Reactor Experiment (SRE). It was decided to place mass limits on material handling and processing and geometry limits on storage and shipment. The absence of moderating material, except in the slug-washing step, is a basic safety feature of the system, with mass or geometry limits being added for the possibility of accidental addition of moderators to the system. It was suggested that storage of nuclear material and cladding scrap in the same area should be avoided if feasible since this would give better control and housekeeping in the potentially hazardous area.

The spread of contamination that may result from the explosion of the NaK bonding when it is exposed to the cell atmosphere was not considered in this report. An adequate review of the hazards involved in the handling of NaK in a contaminated environment should be included in an over-all safety review of the process hazards.

Another ORNL study[23] considered the possibility of accumulating an explosive concentration of hydrogen in the fluoride-volatility-cell

ventilation system, for example, from leaks in the hydrofluorination vessel. No information was presented concerning nuclear safety, specifically, since no potential hazards existed in the given scale of operation.

A new design was proposed for the cell ventilation to reduce the possible hazard of high hydrogen concentration in the ventilation system. The resulting hydrogen concentration from a total rupture of a process vessel (with the redesigned ventilation system) was estimated to be well below the lower explosive limit of hydrogen in air. The possibility of an explosion in such a system when processing highly radioactive material would probably require a very stringent redesign of the ventilation system. For instance, it might be necessary to include an interlocking cutoff of hydrogen fluoride to the process in case of failure of the ventilation fans. (J. C. Suddath)

References

1. J. L. Zegger and G. P. Pancer, APPR-1 Research and Development Program Decontamination Program. Task II. Volume I. Contamination and Decontamination in Nuclear Power Reactors, APAE-43 (Vol. I), Feb. 13, 1959. (Unclassified AEC report.)
2. D. G. Stevenson, Great Britain Atomic Weapons Research Establishment, Aldermaston, May 1958. (Unpublished.)
3. G. P. Pancer and J. L. Zegger, APPR-1 Research and Development Program Decontamination Program. Task II. Volume III. Recommended Procedure for Decontamination of a Stainless-steel Steam Generator, APAE-43 (Vol. III), Feb. 13, 1959. (Unclassified AEC report.)
4. W. H. Snavely and P. J. Manno, Handling Contamination in Spent-fuel Shield Water, *Nucleonics*, 17(3): 121-123 (1959).
5. J. W. Savage and D. J. Stoker, Equipment Decontamination by Abrasive Blasting in the Processing Refabrication Experiment, NAA-SR-3261, April 1, 1959. (Unclassified AEC report.)
6. A. E. Symonds, Jr., Removal of Tritium Contamination from Surfaces of Metals, DP-367, April 1959. (Unclassified AEC report.)
7. J. Bellier and M. Tourasse, Concrete Pressure Vessels, *Nuclear Energy Engineer*, 13(130): 150-151 (March 1959).
8. F. T. Binford and T. H. J. Burnett, A Method for the Disposal of Volatile Fission Products from an Accident in the Oak Ridge Research Reactor, ORNL-2086, Aug. 2, 1956. (Unclassified AEC report.)
9. R. O. Brittan, Reactor Containment (Including a Technical Progress Review), ANL-5948, May 1959. (Unclassified AEC report.)
10. F. B. Hornibrook, H. Freiferger, and A. Letvin, A Study of Durability and Void Characteristics of Concrete Admixtures, Principally of the Air Entraining Type, *ASTM Bulletin*, 46: 1320 (1946).
11. Kaiser Engineers, unpublished data.
12. Report to AEC Reactor Safeguards Committee Regarding Prestressed Concrete Reactor Safeguard Shell for 3000-kw Dual-cycle Boiling-water Reactor, The Preload Company, Inc., Dec. 16, 1955.
13. B. L. Greenstreet and S. E. Moore, Oak Ridge National Laboratory, unpublished data, July 8, 1959.
14. Letter from J. J. Closner, The Preload Company, Inc., to M. H. Fontana, Oak Ridge National Laboratory, Sept. 8, 1959.
15. A. W. Richards, Report on Trip to Hanford, Washington, on September 24, 1951, Application of Polyethylene Coating to Concrete, NYO-3388, Dec. 6, 1951. (Unclassified AEC report.)
16. C. R. Binner, C. B. Wilkie, and Philip Miller, High-density Concrete Shielding, Feb. 15, 1949; Supplement: Heat Testing of High-density Concrete, HKF-1 and Supplement, June 1, 1949. (Declassified AEC report.)
17. A. E. Reeves, Great Britain Atomic Energy Research Establishment, Harwell, 1955. (Unpubpublished.)
18. Steel Construction Manual of the American Institute of Steel Construction, AISC, New York, 1947.
19. ASME Code, Section VIII, Unfired Pressure Vessels.
20. Norman Ketzlach, Nuclear Safety in Processing Less Than 5.0% U^{235} Enriched Reactor Fuels, HW-58049, Dec. 11, 1958. (Unclassified AEC report.)
21. C. W. Pierce, Reprocessing of Power Reactor Fuels. The Enrico Fermi Fast Breeder Reactor Fuel Progress Report No. 2, BNL-549, March 1959. (Unclassified AEC report.)
22. J. C. Suddath, Oak Ridge National Laboratory, 1959. (Unpublished.)
23. S. H. Stainker, Oak Ridge National Laboratory, 1959. (Unpublished.)

Contamination of the Lithosphere

One aspect of hazards evaluation which has been neglected in the past has been that associated with the contamination of surface and subsurface soils. The problem of ground contamination has, in general, been avoided through a judicious choice of assumptions. Hazards summary reports, in particular, have utilized this expediency, as evidenced by the many assumptions of the direct and uninhibited movement of released activity into surface waterways. To justify these assumptions, it has been customary to further assume that the release occurs during times when the ground is frozen or that subsurface channeling either exists or occurs concurrently. These assumptions are usually permitted because they result in pessimistic conclusions as to the amount of activity in the water, and thus they are conservative.

As the number and size of nuclear installations increase and their degree of isolation decreases, it becomes more apparent that overpessimism is a luxury that can no longer be afforded. A more realistic treatment of the ground contamination problem will be required to properly account for the effect that the environment would have in the ultimate disposition of released activity. The problem is far from simple, one reason being that many disciplines are involved. These could include, in addition to geochemistry and geophysics, both ground and surface-water hydrology, as well as such apparently unrelated fields as limnology, soil science, and agronomy. In view of this diversification, it might be expected that the published accounts of the problem of ground contamination would be limited to only one of its phases.

Although this has been the case, in general, several isolated examples of treatments of a more comprehensive nature which purport to be complete and conclusive have appeared in the literature. The temptation to draw conclusions about the over-all problem from a detailed study of one of its parts has been partially responsible for the appearance of such treatments. It is likely that these will continue to appear until the complexities involved are better recognized and the proper talent is brought to bear on the problem.

The groups that already appreciate these complexities are those which are associated with the various aspects of waste disposal. The wide diversification of talent commonly found within these groups is direct evidence of such recognition. In fact, much of the present-day knowledge in the broad field of lithospheric contamination has come directly from the various research programs in the field of waste disposal. Even among these groups, agreement is not always reached. The following review is one which illustrates the variations in interpretations which are commonly found among these groups.

Direct Disposal to the Ground

The Savannah River Plant[1] routinely disposes of large volumes of low-level liquid radioactive waste to the ground through open seepage pits. Prior to December 1957, only several hundred curies of nonvolatile fission products had been released,[2] an amount too small to constitute a hazard to the local water resources because of adsorption on the soil. However, the report cited in reference 1 suggests that the movement and distribution of the liquid waste and its fission products can be mapped and monitored far more accurately than seems possible.

One of the illustrations in the report shows contours of the water table under one group of seepage basins and the direction of movement of waste, vertically and laterally, from the seepage pits to mounds on the water table. Clay lenses and perched water tables are responsible for the lateral movement in the unsaturated zone above the water table. However, somewhat different water-table contours could be drawn to fit the observed data, and different conclusions could be drawn as to the direction of movement of water and waste, both in the zone of aeration and below the water table. In fact, the main water-table contours shown in Fig. 1 of reference 1

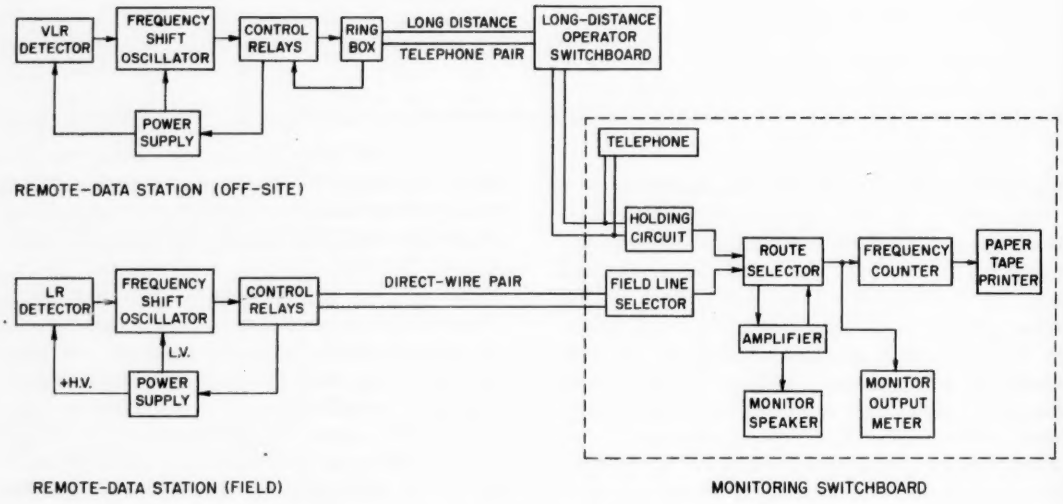

Figure 1—Block diagram of remote radiation-monitoring system developed by NBS.

could be the result of a different flow pattern of the waste above the water table. Movement below the water table is only slightly less uncertain, and therefore the interpretation and reliability of data from water monitoring wells is open to question. The zones of most rapid movement and highest fission-product concentration may have been missed since the source of the radioactivity in any particular sampling well is not always certain.

The report suggests that geologic correlations substantiate the particular interpretation made. The sediments involved are lenticular deposits of gravel, sand, silt, and clay with little continuity and, apparently, no key beds; consequently, a positive correlation, even over distances of a few hundred feet, is not possible. Only the texture of the beds was considered in making the correlations between wells. The beds described in the well logs were divided into 11 categories as a result of a description based on visual inspection, and then relative permeabilities were assigned to these categories. For example, medium sand, which is highly permeable, is number 2, while medium sandy clay, which is very much less permeable, is number 7; in borderline cases, such a characterization would be arbitrary and would vary from observer to observer. It is not possible to divide underground beds into categories representing permeability by visual examination of well samples, nor is it true that "porosity and permeability gradually increase as the grain size of the sediment increases." Plotting such data for the purpose of defining the paths of underground flow in material as lenticular and variable as that indicated in the report is not warranted. The stratigraphic correlations shown do not carry conviction since too few units can be identified from well to well. The geological and hydrological data do not provide an unequivocal pattern of waste movement.

The report states that "saturated soil through which ground water moves is less effective in removing radioactivity than is dry soil through which ground-disposed wastes percolate." The first reason given is that "saturated soil has no storage capacity for additional liquids." However, water held in the capillary pores above the zone of aeration is just as truly in motion as water below the water table, particularly under a seepage pit where there is a constant flow of moisture from above. One is not fixed and the other mobile; the question is one of relative rates. This leads to the second reason for discounting the possible benefits of delay and decontamination below the water table because, it is stated, "water moves through water-saturated soils much faster than through dry soils." This, however, depends on many factors, one of which is the driving-force gradient, which is the full weight of the water above the water table but which will be a small fraction of the weight below the water table. Rates of ground-water

movement in formations like those at the Savannah River Plant are commonly of the order of less than 1 in. to as much as 1 ft per day, whereas rates of movement of vadose water may vary from motionless to many feet per day. Thus the authors' statement regarding relative movement of water through dry and saturated soils is not valid, particularly since it refers to the soil under the waste pits which, although it is not saturated, is far from dry. It might be added that the horizontal distances traveled below the water table are very much greater than the theoretical distances traveled above the water table; therefore, travel time and adsorption could well be far greater below the water table than above. The final argument that "soils that are saturated with ground water are already saturated with the ions present in the ground water, whereas dry soils generally have greater unused ion-exchange capacity" suggests that there are no vacant positions in the dry clay, which is not correct. The mechanism by which activity is held up is one of ion exchange, where an ion of radioactive cesium, for example, is substituted for an ion of potassium.

Subsurface Contamination from Nuclear Explosions

A source of subsurface contamination, other than accidents involving activity releases or the deliberate disposal of radioactive waste into the environment, is that resulting from the detonation of nuclear explosives at or beneath the surface of the ground. Various peacetime applications have been proposed for the use of such explosions which take advantage of one or the other of the properties of nuclear detonations (Project Plowshare). Common to all such explosions is the production of fission products, which depend solely on the fission yield, as well as some amount of induced activity, which depends on the number of neutrons escaping the assembly, as well as the type of medium into which these neutrons are absorbed.

A series of reports dealing with various aspects of Project Plowshare has recently been made available. One of these reports describes in some detail the amounts and types of radioactivity associated with the underground explosions.[3] The subject is introduced by a brief description of the explosion phenomena. The report points out that the extreme temperatures reached immediately following detonation completely vaporize all the material in the immediate vicinity and that the individual chemical properties of the materials are not of importance. Condensation of the less volatile components then begins to occur, with the condensates being bound within the molten layer at the walls of the cavity. Venting or collapse of the cavity during these early times determines whether appreciable amounts of the activity escape from the molten zone.

It is customary to measure the energy release in terms of "kilotons," a unit which the report defines to be 10^{12} g-cal. Based on a *prompt* energy release from the fission process of 179 Mev, this corresponds to 1.46×10^{23} fissions or the complete fission of about 55 g of the fissionable material. In terms of the production of one of the more important fission products, 1 kt of fission yield produces about 1 g of Sr^{90}. Assuming that the yield-mass curve for the fission process in nuclear explosions is the same as that in a thermal reactor, this would correspond to about 2750 Mwd of reactor operation. This assumption is presumably made in the report since the yield-mass graph shown is that for thermal fission. This assumption possibly results in overestimates of strontium production since the peaks of the yield-mass curve tend to decrease as the energy of the fissioning neutrons increases. Conversely, the production of isotopes in the trough would be underestimated because the trough tends to fill up as the neutron energy increases.

Both the abstract and the introduction of the report cited in reference 3 clearly indicate that it is concerned with the hazard aspects of underground nuclear explosions; in particular, Sr^{90} and Cs^{137} are mentioned. Both these isotopes are long-lived and would possess some degree of hazard potential for several centuries, and yet the report discusses their relative contribution to the total radioactivity for the period of time between 10 sec and 1 hr after the detonation. Since the discussion is relative to the fraction of activity escaping via venting, presumably this would explain the lack of completeness in the data presented.

The report also describes a series of measurements made on debris from the site of several underground explosions at the Nevada Test Site (NTS). These measurements, which show the fraction of various fission products found in the fused material, rubble, or vented debris, would only be valid for underground explosions

of the same character and in the same geological structure.

The problem of ground-water contamination is also discussed, particularly with respect to the flow rates of the ionic constituents. This material presented was taken directly from another report in the Plowshare Series, which is reviewed later in this section.

The report[3] concludes with an interesting discussion of the radioactivity induced in the material surrounding the explosion by the leakage neutrons. Based on an assumption of leakage of 1 neutron per fission, the activity produced in a "typical medium" is calculated. By assuming the typical medium to contain 20 wt.% water, the greatest fraction of the escaping neutrons is eliminated by hydrogen capture. Those neutrons not captured by hydrogen are then absorbed by the various elements in the surrounding materials in accordance with their contribution to the total macroscopic cross section.

The report lists Na^{24}, Al^{28}, Mn^{56}, Fe^{59}, and Co^{60} as the important radioactive nuclides and then proceeds to calculate the activity of each that is produced per kiloton of fission yield. A "new" term is also introduced, the "relative biological effectiveness," which is defined as the product of the activity in curies and the total gamma-ray decay energy of the isotope in question, and thus represents the power developed by the particular isotope at the time of its formation. Actually, the relative biological effectiveness (RBE) is a time-honored concept and is quite familiar to those working in the various phases of radiation protection.[4]

By the misconstrued use of the RBE, the isotope Al^{28} is said to be the most important. Actually, this isotope, because of its half life of 2.3 min, would disappear long before it could contribute to any hazard. Somewhat more noteworthy is the omission of several isotopes which could make a significant contribution to the hazard. Of these, perhaps Ca^{45} is the most important. This isotope would be produced in the typical medium at a concentration of about 100 curies/kt and, because of its low maximum permissible concentration, would represent an important contribution to the hazard potential from induced contaminants for several years following the explosion. Undoubtedly, other components of the induced activity more important to the overall hazard could be found if the search were extended to include the beta emitters.

(L. C. Emerson)

1. *Movement and Depletion of Ionic Species in Ground Water.* The movement of the ionic constituents within the ground-water system mentioned previously is described in more detail in another Project Plowshare report.[5] The equation relating the rate of flow of ionic species (F_a) is given by

$$F_a = \frac{F_w}{1 + K_d \rho}$$

where F_w is the flow rate of water within a particular aquifer, K_d is the distribution coefficient for a specific element, and ρ is the ratio of the weight of mineral to the volume of water per unit volume of aquifer material.

By assuming, say, a ρ of 1 and a flow rate of 10 ft/day, the average flow rate for any ion can be calculated if the value of K_d is known. These values are listed in the report[5] for several nuclides. Magnitudes of flow rates would depend on the selection of values of K_d; hence, if pure Colorado bentonite is the medium, a K_d of 2470 is reasonable, as given in the report. On the other hand, for a more sandy medium, this value may be too high.

Since experimental data exist for Hanford soils, it seems appropriate to consider the range of K_d values reported in the literature for strontium. The value of 416 listed for Hanford composite soil was derived from the data of reference 6. A value of 70 for Hanford composite soil was later reported by McHenry.[7] Although the statement is made[5] that varying the *p*H has a very small effect between *p*H 2 and about 9 on the K_d for strontium, it is felt that this is a misinterpretation of the data of reference 6, which show that, within the range studied, the extractability of Sr^{90} is approximately constant; this analysis was made in a system where the solvent contained 1 mole of acetate ion (rather than in a distilled-water system of the type normally used for the determination of K_d's). McHenry[7] reports that *p*H has a very marked influence at *p*H values less than 8.0. Rhodes[8] reports that for Hanford soil the K_d is 4 at *p*H 5.0 and that it increases to 130 at *p*H 11.0 (the soil was calcium saturated). If a K_d of 4.0 is used (the other extreme situation, in contrast with a K_d of 2470), the flow rate of strontium becomes 2 ft/day (in contrast to 0.004 ft/day), assuming a 10 ft/day flow rate for water.

In addition to a consideration of the average rate of flow of ionic materials, the report[5] discusses the depletion of the available ions within the ground water during such flow. The calculations given are based on a stepwise determination of the activity in progressive zones within the system, with the quantity of activity in any one zone being based on that in the preceding zone. The procedure is quite cumbersome, but the results are presented in the form of a graph which uses zones and time cycles in a fashion which permits their adaptation to any specific case. The report gives a general equation to represent the fraction of the initial activity present in any zone n after any number of cycles c. The equation in the report contains two errors and should be written:

$$A_n^c = \left(1 - \frac{1}{K_d}\right)^{c-n+1} \left(\frac{1}{K_d}\right)^{n-1} \binom{c}{n-1}$$

where K_d is the distribution coefficient and $\binom{c}{n-1}$ is the the binomial coefficient. The binomial coefficient in the original expression is in error, as is the definition of the binomial coefficient given in terms of factorials. The correct binomial coefficient and its correct factorial expression are

$$\binom{c}{n-1} = \frac{c!}{(c-n+1)!\,(n-1)!}$$

Other complicating factors can be introduced, including the nature of the ground water, to show that strontium movement is sensitive to differences in dissolved ions in distilled tap water.[9] Another significant factor is the nature of the material in which detonation occurred. If the ion-exchange material is similar to quartz, then there is a temptation to ask what the flow rate of strontium will be in this medium. The flow rate of plutonium was calculated in the report[9] to be as large as 1.5 in./day; and the statement is also made that the order of decreasing K_d is given as Ru < Sr < Cs < Y < Pu ~ Ce. Further, Christenson et al.,[9] working on Nevada tuff material, made the statement that plutonium is readily retained by the tuff and that, at Los Alamos, plutonium in waste discharged at the ground appears to remain at the point of discharge. With reference to strontium, the report states that "Sr^{90} is not retained by the tuff nearly so well as cesium and plutonium, and it is much more easily released."

The data from different sources, therefore, do not substantiate the conclusion that the "ion-exchange behavior of the major fission-product radioactivities is well understood and that their behavior serves to essentially eliminate the potential hazard from ground-water contamination with debris produced by an underground nuclear explosion." In fact, what is needed is more data and a better understanding of the ion-exchange behavior of the fission products. (T. Tamura)

2. *Effect of Nuclear Explosion on Immediate Surroundings.* Section I of the Project Plowshare report[5] discussed previously is also concerned with "The Nuclear Explosion and Its Effects on the Immediate Surroundings." The essence of this first part of the report is that (1) an underground explosion, if contained, will briefly melt the rock around it; (2) the radioactive materials resulting from the explosion will be taken up by the melt; and (3) the radioactive materials will be permanently held by the glass that will form as the melt cools. This is substantially what happened in the case of two contained underground explosions at NTS. The volcanic tuff in which these detonations took place is an unusual rock; it has a porosity of 10 to 35 per cent but very low permeability. The thickness of the rock cover required to contain an explosion, as claimed, may be relatively independent of the strength of the rock; the shock wave travels so fast that inertia alone is the controlling factor. But the physical and chemical nature of the rock have much to do with the formation and continued brief existence of a molten-glass-lined cavity that is essential for the incorporation of the activity in the glass, and the peculiar properties of the Nevada tuff were well adapted to this need. There is no assurance that the same, or even comparable, results would be obtained in oil shale, sandstone, or granite; the report states that no glass would be formed in limestone, although there is no need to take measures to prevent the movement of any release activity "because of the nature of water flow in limestone." What is favorable about the nature of water flow in limestone is not made clear; geologists regard limestone as one of the most dangerous aquifers for the movement of contamination.

It is also suggested that the rock just outside the melted zone will be crushed to a very fine

powder that will be impermeable; but, in both the test shots, the cavity collapsed after a minute or two and formed a chimney of coarse broken blocks which, in a moist climate, would have filled up with ground water. It is stated also that the leaching tests on the glass from the Nevada tests show that earlier work on the slow leaching of other glasses applies, in general, to the leaching of rock fused by a nuclear explosion; however, the investigators who performed these tests are not willing to make such generalizations.[10] The earlier work referred to was a study for which the data extended to four months of contact leaching; and, in the report, these data are extrapolated to imply very little leaching in the next few centuries. The term "insoluble" should not be construed to mean permanent insolubility. Very little is known about the chemical stability of the glasses; short-term leaching of materials may represent diffusion losses, and not decompositional release.

3. *Occurrence and Flow of Underground Water.* The section of the report[5] on the occurrence and flow of underground water is said to be summarized from Tolman[11] and Meinzer,[12] but the condensation is not entirely accurate. One example will serve to illustrate the type of error which has been made. If the water table in an aquifer is lowered, as by pumping, some of the water remains held by capillary forces in the smaller openings above the water table, and it will remain fixed there permanently, except as it may move into the vapor phase. The proportion so held is called the "specific retention," and in a medium-grained sand it may be equal to one-tenth of the total volume of the aquifer, or one-third of the total water present when the sand is saturated. The author, however, has the impression that, in the saturated material below the water table, the same proportion of the water is held immobile but is able to exchange with, and thus dilute, any contaminated water that might move past. In fact, it is suggested that this would retard the movement of a contaminant as much as does ion exchange. However, the capillary forces that bind the water held by specific retention only come into existence as draining of the sand produces air-water interfaces, which do not exist below the water table. Therefore this particular safety factor does not exist.

Radiation Monitoring of Large Areas

Aerial Surveying

The routinely available monitoring equipment is inadequate to measure the distribution of contamination following an accident or other event that releases contamination over a wide area. During such times it is necessary to resort to means that provide the essential data with a minimum of effort. The use of the airplane proved to be an effective aid in making rapid surveys of the country-wide contamination following the Windscale[13] incident on Oct. 10, 1957.

The aircraft used for the survey was an Austin "Autocar 145," a two-seated aircraft capable of carrying 200 lb of equipment at about 100 mph, with a flight endurance of 4 to 5 hr. The equipment installed included a recording scintillation counter with a NaI(Tl) crystal $4\frac{5}{8}$ in. in diameter, and 1 in. thick) and a 35-mm air-survey camera for position plotting. The results of the surveys were plotted on radiometric profile charts, with the flight lines shown separately on reproduced maps of the area.

When flying at a height of 500 ft above the ground, the coverage of the aircraft was a strip approximately 1000 ft wide. Not only was this about 25 times the ground area per line-mile that could be surveyed by a car-borne instrument but also the results were independent of the local "shadowing effects" of trees, buildings, and thermal currents over roads. The airplane measurements are dependent, in part, on the altitude being maintained reasonably constant; however, recording altimeters would allow corrections to be made. The contamination of the airplane is not as much a problem as would be the case with car-borne equipment. The operation of the aircraft, however, is limited by weather and lighting conditions, and the determination of the aircraft position is difficult.

Careful preplanning of flight patterns, as well as background flights prior to an incident, is highly desirable since the background varies widely over different geological formations. It has been reported that this technique of aerial surveying is useful when the ground contamination delivers a dose rate of about 75 μr/hr at a height of 1 meter above the surface.[14] For Cs^{137} this would correspond to about 8 $\mu c/m^2$. Un-

doubtedly, with better knowledge of the background, this performance could be improved.

(F. J. Davis)

Remote Telemetering

Another monitoring technique is available for large-area coverage which has the advantage of providing essentially continuous, as well as much more quantitative, data. The remote-telemetering system referred to here has the disadvantage, however, of providing only spot coverage. One such system, developed by the National Bureau of Standards (NBS)[15] at the request of the Instruments Branch of the AEC Division of Biology and Medicine, was field tested[16] in the measurement of radioactive fallout from the 1957 atomic tests at NTS. The present instrumentation has evolved from previous direct-wire and radio-controlled telemetering systems used during other similar tests over the past eight years.[17–19]

A remote system offers multiple advantages for radiologically monitoring areas that might become contaminated by fallout, accident, mishap, and/or induced activity, including reduction of personnel exposure and of time required for assessing the degree and contours of radiation levels immediately following and after the incident. Any particular application would dictate the relative weight given factors of reliability, accuracy, versatility, bulk, weight, and speed, as well as the economics of construction, operation, and maintenance. This system collects telemetered radiation data locally by means of directly coupled field lines up to 15 miles long and off-site by commercial telephone circuits from detectors in communities as far distant as 330 miles. The immediate objectives in the development of the system were to (1) monitor residual gamma-activity levels after the release of activity along planned routes by early recovery parties investigating test structures near the release point, (2) monitor off-site fallout intensities and patterns, and (3) evaluate equipment operation limitations, future design-change criteria, and relative reliability of telemetered radiological data using comparative techniques.

Functionally, the system consists of one or more receiving and control switchboards interrogating multiple remote-data stations through either direct-wire field lines or commercial telephone-circuit pairs, as shown in Fig. 1. The called direct-line field station, actuated by a d-c polarized relay, completes a stepping-switch-controlled cycle of information and resets. By comparison, the remote long-distance telephone access station "ring box" automatically "answers" the long-distance operator's dialing (voice-requested by the central-station operator), transmits a similar cycle of coded data, automatically terminates the call, and resets. Normally, two direct-line stations share a common pair of wires; additional stations may be added on a stepping-switch time-sharing basis. Likewise, the number of telephone stations is limited only by the number of available circuits and equipment.

The remote-data-station gamma-ray detector (ion-chamber-operated halogen Geiger-Mueller (G-M) tube surrounded by 0.012 in. of lead for energy compensation) produces, across an integrating circuit, a d-c voltage that is an approximate log function over three decades of the incident radiation intensity. Depending on the application, either the VLR detector (0.3 mr/hr to 100 mr/hr) or the LR detector (10 mr/hr to 30,000 mr/hr), or a combination of these detectors is used. In turn, this varying output voltage shifts the output frequency of a voltage-sensitive local oscillator from 800 to 1400 cps. This oscillator-output frequency, telemetered over the direct lines or telephone circuit, is received at the monitoring central station, read on a frequency counter, and then printed on paper tape. The radiation intensity at the remote station is determined from the tape by comparison with a previous calibration curve using Co^{60} calibration sources, with an estimated overall accuracy of 20 per cent. The energy- and directional-response characteristics of the detectors, as well as the oscillator-calibration curve, are required to interpret the telemetered data. One complete data cycle reporting radiation intensity, station identity, and calibration check points requires 100 sec per station challenged.

The remote units, self-powered by 16 dry batteries, are energized only when challenged. Battery life was sufficient for the duration of the test series[16] for long-distance telephone-line stations; local stations frequently challenged required replacement. Where frequent relocation of remote-data stations was necessary, their bulk and weight resulted in considerable abuse from handling procedures. Packaging redesign was recommended on the basis of enviromental factors (dust, blast-wave

shock, etc.), and minor electronic redesign was necessary to correct oscillator dependence on filament battery voltage and tube characteristics.

Subsequent advances in semiconductors, components, construction, and packaging techniques would permit considerable improvement in this system. Telemetering direct-wire systems based on integrated G-M tube current detectors have appeared elsewhere recently. Lockheed[20] and Tracerlab[21] have announced transistor-amplified d-c telemetering systems using single line pairs; a system proposed at ORNL[22] transmits a-c power and returns the d-c signals over single telephone pairs from a central, monitoring, combined display board and power supply.

(J. A. Harter)

Personnel Exposures Following Accidental Releases

One of the problems associated with the operation of liquid-moderated reactors is that of the local contamination associated with the neutron-induced activities of impurities within the moderator or even of the moderator itself. Heavy-water moderators are particularly troublesome in this respect because of the long half life of the resulting activity (12 years), as well as the large contamination potential due to its relatively high vapor pressure. The activity produced is tritium, which results from an (n,γ) reaction on the deuterium. Because of its long half life, the concentration continues to increase for essentially the full lifetime of the reactor.

An account of the potential hazards associated with the operation of a typical heavy-water-moderated reactor has recently become available.[23] In addition to the hazards, this report quantitatively discusses the origin of the tritium contamination, as well as the maximum permissible concentrations under a variety of exposure conditions.

A calculated value for the saturation activity of tritium was obtained using nuclear parameters available from several years of operational experience. The value obtained, 16 mc/ml of D_2O, compares with a value of 9.4 mc/ml obtained from the initial slope of the measured build-up of tritium plotted vs. the time integral of the reactor power. Extrapolation to 10 years of operation yields concentrations of 4 mc/ml with a total tritium inventory of 4×10^4 curies.

Using relevant data on the standard man, calculations are made of the integrated dose for a single intake, i.e., the total dose delivered to the body throughout life from a single intake and an estimate of this dose based on the concentration of tritium in urine following such an intake. The results of the single intake calculation are shown in Fig. 2. The dose at equilibrium from

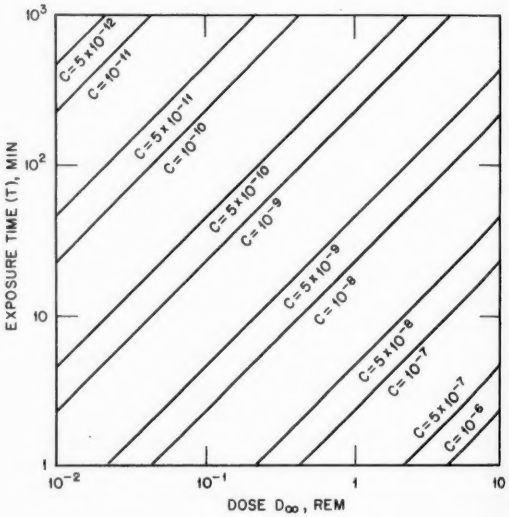

Figure 2 — Total dose received from inhalation of an atmosphere containing HTO for T min as a function of concentration C (curies/cm^3), assuming no elimination of tritium during time T (biological half life of tritium is 12 days).

long-term continuous exposure at a constant level is also calculated, and the $(MPC)_a$ value so obtained points out an error in the $(MPC)_a$ values for DTO in NBS Handbook 69.[24] The chairman of NBS Committee II indicates that the error is to be corrected, along with some other changes in an addendum that is to appear shortly.

A British report[23] also discusses the maximum hazard to be expected from saturating the DIDO reactor hall with contaminated D_2O. The exposure times resulting in 10-rem doses are shown in Table V-1 for various levels of contamination and biological half lives. The exhaustion of the contaminated atmosphere in the normal ventilation system is shown to represent no unacceptable hazard, even under pessimistic meteorological conditions.

(L. C. Emerson and W. S. Snyder)

Table V-1 TRITIUM CONCENTRATION OF AIR IN DIDO REACTOR HALL WHEN SATURATED WITH TRITIATED D_2O VAPOR

Tritiated D_2O vapor concentration, mc/g	Tritium concentration, curies/cm^3 $\times 10^{-6}$	Exposure time (for D_2O = 10 rem), min	
		T_b = 12 days*	T_b = 5 days*
1	1.7	140	340
2	3.3	70	170
3	4.9	48	114
4	6.6	35	85
5	8.3	28	67
10	17	14	34

*Biological half life of tritium.

During the few weeks following the Windscale accident, it was observed that the gamma-ray activity of certain persons in the Surrey region was higher than that which could be accounted for by the known activities.[25] This increase in activity was subsequently traced to I^{131} by detailed measurements on a number of persons. The measurements were made in a low-background laboratory using an NaI(Tl) crystal, 2 in. thick and 1.75 in. in diameter, optically coupled to a photomultiplier tube, the output of which was fed via amplifiers to a 100-channel pulse-height analyzer.

Of the 18 persons measured, 16 were residents of the greater London area, 1 was from Belmont, and 1 was from Leeds; 16 were adults, and 2 were children (ages 10.5 and 12 years). Each individual was counted for 15 min. Most subjects were counted only once, and a series of reliable measurements was made for just one individual. Because these measurements showed an elimination rate from the thyroid that was slower than the radioactive-decay rate of I^{131}, it was realized that the uptake of I^{131} was continuing. Since the atmospheric content was negligible, another source of contamination was sought; it was found to be the milk supply.

By making several assumptions, the total mean radiation dose to the thyroid of the individual was calculated to be 0.13 rad, and the dose for the average person in the Surrey and London groups was estimated to be one-third this value, or 0.04 rad. The two children and one adult had the same milk supply and were drinking the same quantity daily. The similarity of results for these three individuals suggests that there is little difference in the uptake of I^{131} into the thyroid for different ages. However, because of different thyroid weights, the total radiation doses to the children's thyroids were about twice the dose to the adult's thyroid.

(M. J. Cook)

References

1. S. O. Reichert, Geology and Hydrology for Disposal of Radioactive Wastes to Ground at the Savannah River Plant, DP-341, December 1958. (Unclassified AEC report.)
2. R. E. Brown et al., Experience in the Disposal of Radioactive Wastes to the Ground, Hearings Before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, January–February 1959.
3. R. E. Batzel, Radioactivity Associated with Underground Nuclear Explosions, UCRL-5623, June 23, 1959. (Unclassified AEC report.)
4. Report of the International Commission on Radiological Units and Measurements, National Bureau of Standards Handbook 62, U. S. Department of Commerce, April 1957.
5. G. H. Higgins, Evaluation of the Ground Water Contamination Hazard from Underground Nuclear Explosions, UCRL-5538, Apr. 8, 1959 (unclassified AEC report); also in *Journal of Geophysical Research*, 64: 1509-1519 (October 1959).
6. Hideo Nishita, B. W. Kowalewsky, and K. H. Larson, Fixation and Extractability of Fission Products Contaminating Various Soils and Clays: I. Sr^{89}, Sr^{90}, Y^{91}, Ru^{106}, Cs^{137}, and Ce^{144}, UCLA-282, 1954. (Unclassified AEC report.)
7. J. R. McHenry, Ion-exchange Properties of Strontium in a Calcareous Soil, *Soil Science Society of America, Proceedings*, 22: 514-518 (1958).
8. D. W. Rhodes, Waste Characteristics Governing Fixation in Soils, in "Report of Working Meeting on Fixation of Radioactivity in Stable, Solid Media," held at the Johns Hopkins University, June 19-21, 1957, TID-7550, pp. 1-3, March 1958. (Unclassified AEC report.)
9. C. W. Christenson et al., Soil Adsorption of Radioactive Wastes at Los Alamos, *Sewage and Industrial Wastes*, 30: 1478-1489 (1958).
10. W. D. Bond, Oak Ridge National Laboratory, personal communication.
11. C. F. Tolman, "Ground Water," McGraw-Hill Book Company, Inc., New York, 1937.
12. O. E. Meinzer, ed., "Hydrology," McGraw-Hill Book Company, Inc., New York, 1942.
13. D. Williams, R. S. Cambray, and S. C. Maskell, An Airborne Radiometric Survey of the Windscale Area, October 19-22, 1957, AERE-R-2890, December 1957. (Unclassified British report.)
14. A. C. Chamberlain, Relation Between Measurements of Deposited Activity After the Windscale

Accident of October 1957, AERE-HP/R-2606, September 1958. (Unclassified British report.)
15. Louis Costrell, Remote Radiation-monitoring System, National Bureau of Standards Reports NBS-4148, NBS-5124, and NBS-5471.
16. S. C. Sigoloff and H. M. Borella, Remote Radiological Monitoring, Operation Plumbbob Report, WT-1509, November 1958. (Unclassified AEC report.)
17. Louis Costrell, National Bureau of Standards, 1952. (Unpublished.)
18. R. W. Johnston, U. S. Atomic Energy Commission, 1953. (Unpublished.)
19. R. W. Johnston, Utilization of Telemetering Techniques in Evaluating Residual Radioactive Contamination, Operation Teapot Report, WT-1182, November 1956. (Unclassified AEC report.)
20. J. L. Bracken, Lockheed Aircraft Corporation, Georgia Division, private communication.
21. H. A. Brown and J. V. Rogers, Tracerlab, Inc., Halogen Tube Remote Area Monitoring System, paper presented at Health Physics Society Annual Meeting, Gatlinburg, Tenn., June 1959.
22. J. M. Garner, Jr., and R. L. Bradshaw, Oak Ridge National Laboratory, Radiation Monitoring Utilizing Average Current from Halogen-quenched G-M Tubes, paper presented at Health Physics Society Annual Meeting, Gatlinburg, Tenn., June 1959; also, private communication.
23. R. M. Fry, The Biological Hazard of Tritium with Special Reference to DIDO, AERE-HP/R-2788, January 1959. (Unclassified British report.)
24. Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, Recommendations of the National Committee on Radiation Protection, National Bureau of Standards Handbook 69, U. S. Department of Commerce, June 5, 1959.
25. G. Maycock and J. Vennart, Iodine-131 in Human Thyroids Following the Windscale Reactor Accident, *Nature*, 182: 1545-1547 (1958).

Nuclear Incidents

For the purposes of this Review, a nuclear incident is defined as a nuclear accident in which one of the following occurs: (1) inadvertent personnel exposure, (2) unanticipated contamination of the surroundings, or (3) significant damage to the reactor.

One of the two incidents described here, the Heat-transfer Reactor Experiment No. 3 (HTRE-3) excursion, resulted in the melting of some of the reactor fuel elements. No personnel exposures resulted from this incident; but, since the HTRE-3 is a direct-cycle reactor, significant area contamination did ensue. Postincident analyses have indicated that the HTRE-3 incident might have been avoided if accepted practices in safety-system design and reactor operation had been followed more closely.

The second incident reported here, the Y-12 accident, resulted from an inadvertent critical assembly in an enriched-uranium processing plant. The accident resulted in serious, but not fatal, exposure to a number of plant personnel. Although the accident may have been avoidable, it was an outgrowth of the design of the facilities and the prevailing nuclear safety criteria.

HTRE-3 Excursion

The HTRE-3 attained criticality at the National Reactor Testing Station on Oct. 24, 1958, after which extensive tests were undertaken, including evaluation of the controls, to provide information for safe operation of the system. This series of tests was terminated abruptly about four weeks later, following a power excursion that melted several fuel elements and resulted in the release of a large quantity of fission products. The chronological events preceding the incident are described here, and the subsequent conclusions of the investigating groups are analyzed.

The reactor was operating with greatly reduced coolant flow so that measurements of heating rates could be made. Several of the chambers, including one of the three safety chambers, had been replaced with heat-rate sensors. Signals from either of the two remaining chambers would, however, produce a scram. During the afternoon of November 18, the reactor was operating at 60 kw, and the reactor behavior was as anticipated. At the conclusion of this run, preparations were made for repeating this test run at 120 kw, the highest power up to that time. Since all ionization chambers were inserted to the full-in position, it was necessary to alter circuit constants in the two linear-flux channels in order for the servo to operate over the range of 15 to 150 kw, corresponding to 10 to 100 per cent of full power. The scram level was correspondingly set at 180 kw.

The final run was started, and it proceeded in a normal manner up to the power range. Control was then switched over to the servo to increase the power level on a 20-sec period. Examination of the log N recorder trace indicates that, at this time, the flux level was rising on a 20-sec period. The servo seems to have been switched on too soon because the flux increased stepwise from 4 to 10 per cent of full power. The period recorder trace shows that the recorder pen was being driven at full speed until a period of 8.5 sec was reached, suggesting that a period scram set at a 5-sec period was narrowly averted. The flux was then held constant at 10 per cent of full power for approximately 20 sec, and thereafter it proceeded to rise on a 10-sec period rather than on the 20-sec period that the servo was programmed to produce. This operation continued to 30 per cent of full power, at which time the log N recorder pen "pegged" at full scale, according to plan. The events during this interval are described in a report[1] by the investigating committee which states: "The power of the reactor increased in a manner which appeared to be as expected, although at a somewhat shorter period (about 10 sec) than had been anticipated." It is not suggested that

these events are related to those that followed but rather that they will bear on the conclusions drawn.

The flux was then rising on a 10-sec period, the log N recorder had pegged against its full-scale stop when the reactor reached 30 per cent of full power, and the only nuclear instruments that remained on-scale were the period recorder and the linear-flux recorder. The trace of linear flux is shown in Fig. 3. Either of two uncompensated ionization chambers would actuate the safety circuits, although only one of these supplied a signal to the linear-flux recorder and the servo channel. During the next time interval of approximately 20 sec, the flux continued to rise, with the period recorder indicating very nearly a 10-sec period until the desired power (80 per cent of full scale on the linear-flux recorder) was substantially reached. At this point, instead of holding at constant power, the linear-flux recorder showed the flux level to be dropping rapidly, and the period recorder started to drive downscale. The servo, seeing a negative error signal, continued to withdraw shim rods. This state of affairs continued for approximately 20 sec, at which time the reactor was shut down by means of a

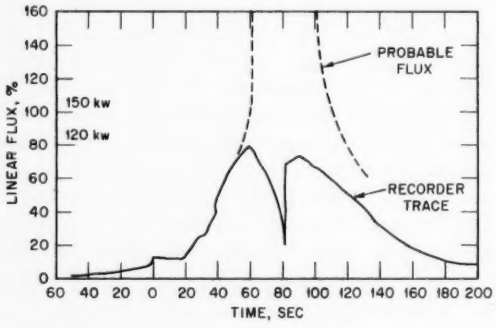

Figure 3—Linear-flux variation.

reactivity loss of more than 2 per cent due to the redistribution of the fuel resulting from melting and/or a temperature scram due to the melting of the thermocouple lead wires. Since the reactor was being operated with reduced coolant flow, the thermocouples were not located in the region of maximum temperature; however, the lead wires did pass through this region, and the circuits were arranged so that, if all the 10 monitored thermocouple circuits were opened, the reactor scram circuit would be activated. As the reactor scrammed, the recorded flux level first increased and then decreased. The operator also scrammed the reactor manually, but his action was preceded (by perhaps 3 sec) by the automatic scram.

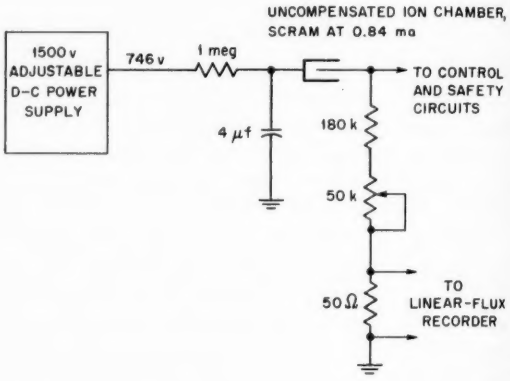

Figure 4—Ion-chamber circuit for power-range indication.

The primary cause of this incident is attributable to insufficient voltage applied to the chamber terminals. The uncompensated chambers were designed to be operated at 1500 volts, but the voltage of the power supply had erroneously been set at 746 volts. In addition, electrical noise filters, which had been introduced during the testing of two previous reactors at the same test stand, had been retained in the chamber power supplies, in the manner shown in Fig. 4. The filter resistor alone would have limited the maximum chamber current to 0.746 ma, whereas 0.84 ma was required to produce a scram.

During the time that the flux was actually rising, the flux recorder showed the flux to be falling, and the servo error signal was in the direction to call for rod withdrawal. Current-limiting resistors were also connected into the compensated chamber-power supplies. In this instance the resistance in series with the gamma portion of the chamber was of a lower value than the corresponding resistance in the neutron-plus-gamma section. Therefore, when excessive currents were being drawn and the current from the gamma section was subtracted from the total, it was logical that a negative period would be indicated when, in fact, the reactor period was positive.

The following conclusions are reported:[1]

It is concluded that the primary cause of the incident was the inability of the linear-flux circuitry to indicate true reactor power. This malfunction is attributed to coincident human factors and not basic design of the reactor or the instrumentation. The failure to remove the filters from the chamber power source and the failure to set the specified voltage (1500 volts) at the chamber power source constitute the human elements. The elimination of either of the above could probably have prevented the incident.

In retrospect, it is conceivable that the incident could have been averted or delayed and damage reduced if the ion chambers had been withdrawn to their design positions of 11 and 14 in. and suitable amplifiers utilized to provide the necessary signal strength.

It is further concluded that the damage to the reactor may have been reduced had the operator chosen to monitor the hottest thermocouples and the scram trip set at a temperature much nearer the expected maximum fuel-element temperature ... for this portion of the planned experiment.

A possible contributory cause was the operator's decision to go to previously unattained powers on automatic servo control. It is believed that the operator might have recognized the malfunction had the reactor power been increased in small steps under manual control.

A reevaluation[2] of the control and safety system has resulted in the following changes:

A separate reactor-flux-measuring system will initiate scram whenever the flux level exceeds a predetermined level (approximately one decade above the level expected in the particular test).

A loss of reactor ion-chamber signal or the existence of an error signal that would eventually saturate the dynamic rod position loop and the shim withdraw bus would elicit an override response.

Nuclear sensor minimum voltage (1200 volts) will cause scram.

A drop to 80 volts in the 115-volt 60-cycle supply will cause override.

Both the conclusions and the remedies are aimed largely toward the prevention or avoidance of this particular incident. Since other accidents may be postulated in which a single event or failure can invalidate an entire safety system, it is the opinion of the reviewer that both the conclusions and remedies should be broadened to include other failures. For example, the conclusion that the operator would have recognized the malfunction if he chose to operate manually is not strengthened by the fact that only a few seconds prior to the incident, on switching to servo control and during the subsequent rise in power, there were two occasions on which abnormally short periods were introduced, and no remedial action was taken.

It is agreed that it was imprudent on the part of the operator to go to previously unattained powers on automatic control. In view of the fact that both the servo and safety systems were operating with chambers that were working at previously unattained currents, it is suggested that it was likewise imprudent to operate the reactor on either manual or automatic control for any purpose other than deliberately to test the safety system. Failure to test under planned conditions must result ultimately in an unplanned test.

The use of only identical types of instrument channels at the new higher power levels means that identical behavior should occur in all channels. Therefore the absence of other instrumentation on range at the designed trip level would cause even a planned test to be hazardous. These facts detract from the conclusion that the basic design was not at fault.

It is a well-recognized principle that safety equipment (for example, pails used for fire fighting) should be used for no purpose other than safety. Since the same chambers were used for both safety and control, it is therefore no coincidence that the safety system failed at the same time as the servo which precipitated the incident.

Monitors to ensure the continued operability of a safety system have been widely used for many years. The absence of these monitors is not recognized in the conclusions, although some monitoring is proposed as a remedy.

(E. P. Epler)

Criticality Accident at the Oak Ridge Y-12 Plant

A radiation accident occurred at the Oak Ridge Y-12 Plant on June 16, 1958, in a facility for salvaging uranium highly enriched in U^{235}. Eight persons were in the vicinity of the critical material, and five received significant exposure. A description of the incident has been published by Callihan and Thomas.[3] The accident resulted from the inadvertent flow into a 55-gal drum of an aqueous solution containing about 50 g of U^{235} per liter of solution. A prompt-critical neutron chain reaction was

established which resulted in about 10^{18} fissions before the system became subcritical.

At the time of the accident, an inventory of the enriched material was in progress, and certain equipment components were being disassembled, cleaned, reassembled, and leak tested for that purpose. Among the equipment were several long 5-in.-diameter pipes used for storing aqueous solutions of U^{235}, the geometry being such as to prevent the establishment of a nuclear chain reaction. The leak-testing procedure employed involved filling the pipes with water from a 55-gal drum (approximately 22 in. in diameter by 33 in. high), examining the joints for leakage, and, subsequently, draining the water back into the same drum.

As a consequence of irregularities in the function and operation of some valves, a quantity of enriched solution was inadvertently transferred from a solvent-extraction system in another area into the 5-in. storage pipe system that was being leak tested; the inventory of the solvent-extraction facilities had been completed several hours earlier, and operations had been resumed. In addition, the enriched solution was transferred into a pipe on which leak testing had been completed. Experimental tests after the incident established the probable mechanism of the accident as the flow of the accumulated concentrated U^{235} solution from the tested 5-in. pipe into the 55-gal drum, followed by the flow of the leak-testing water from other 5-in. pipes in the system. The dimensions of the drum and the concentration of the solution permitted the system to become critical. A volume of 56.2 liters, containing 2.10 kg of U^{235} standing at a height of 23.45 cm, was estimated to be the initial delayed configuration. The reaction was terminated through the continued inflow of the large volume of leak-testing water.

Although it was not possible to reconstruct the reactivity-vs.-time pattern and since the available evidence could possibly be explained in terms of other combinations of events, the following sequence of conditions was suggested. No strong ambient neutron field existed at the scene of the accident (the most likely source being the $O(\alpha,n)Ne$ reaction between U^{234} alpha particles and the oxygen in the water); and, in consequence, this system was prompt critical before any energy was omitted. Once started, the power rose rapidly to a high level and initiated a period of cycling (estimated as 2.8 min), during which fission-produced gases reduced the density and drove the solution subcritical. Subsequently, when these gas bubbles escaped, the system again became prompt critical, with the power level again rising (delayed-neutron source). The inception of a period of boiling caused a sharp decrease in density, and the system settled into an equilibrium condition somewhere in the delayed-critical range, where it was maintained (principally by vapor formation) about 18 min. Estimates of the energy released during the accident, as determined by analyses of samples of the solution found in the 55-gal drum, are given in Table VI-1.

Table VI-1 ESTIMATES OF ENERGY RELEASE DURING ACCIDENT

	Based on sample 1*	Based on samples 2 and 3*
Volume of solution yielding sample, liters	180	252.8
Mass of U^{235} in total volume, kg	2.5	2.4
Total number of fissions ($\times 10^{18}$)	1.3	1.3
Energy release, Mev $\times 10^{20}$	2.6	
Energy release, kw-hr	11	

* Sample 1, taken 8 hr after the accident, contained 14.0 g of U^{235} per liter; samples 2 and 3, which were taken one month after the accident, contained 9.6 g of U^{235} per liter.

Health physics aspects of the accident have been described by McLendon.[4] The initial radiation burst occurred at approximately 2:05 p.m., and sirens of the radiation monitoring system sounded the alarm. Shortly thereafter, intensities of more than 100 mr/hr were detected at a distance of 350 ft from the accident. Within 1 hr after the incident, a survey of the adjacent plant area determined that there was no direct radiation or significant contamination outside the immediate environs of the accident site.

About 3 hr after the incident, a health physics team undertook a survey of the salvage area in which the incident occurred. At a distance of 100 ft from the drum, the radiation dosage rate was 60 mr/hr. The gas-mask canisters of men at this distance for a period of 10 min read 10 to 15 mr/hr, indicating the presence of air-borne contamination. To prevent the spread of air-borne contamination from the salvage area

to other parts of the building, a small fan that exhausted through a CWS filter was utilized to keep the area under a negative pressure. As a safeguard against a reoccurrence of criticality in the 55-gal drum, a scroll of cadmium metal sheet (18 in. long by 14 in. high, and weighing 8.9 kg) was inserted into the drum by manipulation of a 10-ft pipe. The drum was then allowed to remain intact for several days prior to removal of it and the contained solution. "Smears" of the floor indicated the surface contamination was confined to the immediate vicinity of the drum. The highest activity detected on these smears was 250 mrad/hr, direct reading, and 16,000 dis/min/1000 cm^2 alpha contamination. Widespread contamination would have seriously hampered reoccupation and rehabilitation of the area.

After removal of the drum, decontamination of the salvage area was initiated and continued intermittently for the next several days. Within one week after the accident, all uranium-recovery facilities were returned to normal operation.

Initial checking of personnel for evidence of radiation exposure was by means of indium-foil strips, which are included in the security badges of all Y-12 Plant employees. The intent of these foils is not to act as a precise dosimeter but rather to provide a quick positive means of segregating the seriously exposed. This determination is accomplished by measurement of beta and gamma radiations from the radioactive In^{116} isotope which is produced by neutron irradiation of the stable In^{115} isotope in the foil. By this means, 12 of the approximately 1200 persons surveyed were sent to the dispensary within 2 hr after the incident. The maximum badge reading observed was 60 mr/hr. In the five most significant cases, badge readings exceeded 20 mr/hr 2 hr after exposure. Those persons who were sent to the dispensary were checked for personal contamination and interviewed briefly, and their badges were rechecked. Individuals showing evidence of beta and gamma body contamination were scrubbed at the dispensary decontamination facility with soap and water and mild acids. The dosimetry of the accident was reviewed by Hurst, Ritchie, and Emerson of ORNL, and the findings of the review were incorporated in the report of the Accident Review Committee.[5]

Early evidence of exposure of personnel to a large burst of ionizing radiation was obtained from the observations of those present in the immediate vicinity of the 55-gal drum. Three persons saw the blue glow known as "Cerenkov radiation," a fog was observed by one of the three, and an odor was detected by several others. The fog and odor may have been produced by the high flux of ionizing radiation, or the fog may have been an illusion caused by direct ionization on the ocular systems and the odor may have been caused by radiation-produced nitrous oxide commonly associated with ozone. (The fog was possibly the result of fumes from the dibutyl carbitol and nitric acid present in the drum.)

The indium-foil strips in the security badges of the eight more seriously exposed personnel read from 8 to 60 mr/hr shortly after the accident. A Geiger counter placed against the throat of each of these individuals read from 0.2 to 0.5 mr/hr. In addition, radiation was detected in urine and blood samples, which were counted with sensitive scintillation counters adjusted so that gamma rays of energy above 0.66 and 2.0 Mev were indicated on separate registers.

An experiment was carried out on June 18, 1958, in which a burro (chosen because its thickness and size approximate that of man) was exposed to a reactor mock-up designed to give the same neutron spectrum as the accidental excursion. The intent was first to determine the ratio of gamma dose (D_γ) to neutron dose (D_n) and then to determine the relation between blood sodium activity and fast-neutron dose. Measured blood sodium values for exposed personnel and for the burro are as follows:

Employee	Blood sodium (mg of sodium per ml of blood serum)
A	3.2
B	3.2
C	3.2
D	3.1
E	3.2
Burro	3.1

The neutron dose (in rads) to the individuals was determined by means of the relation

$$D_{n_H} = \frac{Na_H}{Na_B} \times 48$$

where the subscript H refers to humans and the subscript B refers to the burro. The gamma dose was determined by the product of this neutron dose and the factors given below:

Employee	Corrected gamma-to-neutron dose ratio, D_{γ_H}/D_{n_H}
A	2.62
B	3.03
All others	2.80

The resulting neutron, gamma-ray, and total doses (in rads) for the exposed individuals are given in Table VI-2.

Table VI-2 Na^{24} ACTIVATION AND DOSE VALUES FOR EXPOSED PERSONNEL

Employee	Na^{24}, $\mu c/cm^3 \times 10^{-4}$	First-collision neutron dose,* rads	First-collision gamma dose, rads	First-collision total dose, rads	Estimated RBE dose,† rem
A	5.8	96	269	365	461
B	4.3	71	199	270	341
C	5.4	89	250	339	428
D	5.2	86	241	327	413
E	3.7	62	174	236	298
F	1.1	18	50.5	68.5	86.5
G	1.1	18	50.5	68.5	86.5
H	0.36	6.0	16.8	22.8	28.8
Burro	2.9	48			

* Gold-foil measurements indicated that the thermal-neutron dose was about 1 per cent of the fast-neutron dose, and thus it could be neglected.

† With an assumed RBE = 2 for fast-neutron dose.

Estimates of the total doses received by the individuals were also made, by independent calculations based on sodium activations, from the indium-foil measurements and from *in vivo* body-counter measurements. Although considerable variation prevailed in the results, the approximate magnitudes and order of the estimates given in Table VI-2 were, in general, substantiated.

Medical information on the accident was compiled by Brucer.[6] Three of the exposed employees (F, G, and H) did not have sufficient doses to require prolonged care, and they were released from hospitalization as soon as it was evident that their radiation exposure was small. Employees A, B, C, D, and E showed decreases in blood elements and other symptoms characteristic of more severe, but sublethal, radiation damage. Mild nausea, accompanied by vomiting, occurred during the first several days following the accident.

Following a three-week period in which the men were almost free of symptoms, blood platelet counts dropped during the fourth and fifth weeks to a level where the possibility of serious hemorrhage complications existed. However, such bleeding did not occur; and, by the sixth week, the men showed unmistakable signs of recovery. Following their release from the hospital on July 30, 1958, a period of long-term observation was initiated.

This accident arose out of a combination of circumstances involving the design of the facilities, the actions of a number of individuals, and the then prevailing nuclear-safety control criteria. Two adjacent processing areas were operated under significantly different approaches to nuclear safety. In the administrative-control area where the accident occurred, principal reliance was on the operating personnel and their immediate supervisors to prevent an accidental nuclear excursion. In the adjacent geometric-control area (the source of the highly enriched uranium solution that formed the critical mass), all processing equipment was designed so that no critical accumulation could occur regardless of other factors, such as the quantity of material in process. At the time of the accident, portions of both areas were in use, and a transition was being made from administrative control to geometric control.

A valved pipeline connected a solvent-extraction system producing concentrated uranyl nitrate in the geometric-control area to the 5-in.-pipe storage system in the administrative-control area. As reconstructed, the solvent-extraction system was in operation at the time the 5-in. storage pipe was undergoing postinventory cleaning. The valve on the trans-area pipeline was inadvertently open for a sufficient period to allow the enriched-uranium solution involved in the excursion to be transferred. Subsequently, the enriched solution was drained into the 55-gal cleaning-water drum, and a critical mass was formed.

To minimize the chances of a reoccurrence of such an event, in so far as is practical, liquid-salvage processing in the Y-12 facilities is now carried out in geometrically restricted equipment. Portable, unsafe containers (such as 55-gal drums) have been banned from the liquid-processing area, communications between various operating and maintenance groups have become more formalized, and the continuous nuclear-safety education program has been intensified. Although the environment in which this event took place and the performance of some individuals might have been improved, a nuclear accident will always be within the realm of possibility whenever potentially critical quantities of fissionable material are being handled. (F. S. Patton)

SRE Shutdown

During a normal power run on the SRE in May, tetralin coolant from the main primary pump shaft seal leaked into the primary sodium system. The reactor was shut down and the pump seal coolant system converted from tetralin to NaK. It was started up again in mid-July to remove tetralin decomposition products through careful low-power operation and cold trapping. Anomalies were noted during the run as expected. When several weeks of operation failed to clear the anomalies, the reactor was shut down to inspect fuel elements.

Since shutdown in late July, all the fuel elements have been removed from the reactor. Twelve of the 43 were found to be severely damaged by high temperature. The mechanism of damage has been established as partial blockage of reactor coolant flow passages by tetralin decomposition products, followed by local overheating of the fuel elements, resulting in formation of an iron-uranium eutectic (m.p. 1340°F) and parting of the type 304 stainless-steel cladding.

Samples of the fuel, cladding, and other components of the fuel assemblies have been metallurgically examined, and coolant and coolant contaminants have been chemically analyzed.

The results of these investigations will be given in a report by the SRE *Ad Hoc* Review Committee scheduled for publication in December. The report will be reviewed in the March 1960 issues of *Power Reactor Technology* and *Nuclear Safety*.

Activities of the Advisory Committee on Reactor Safeguards

A recent AEC publication[7] gives a summary of the proceedings of the Advisory Committee on Reactor Safeguards (ACRS) and of the public licensing hearings held by the AEC from January 1958 to March 1959, inclusive. In addition to a brief summary of the status of reactor projects which were reviewed by ACRS, a letter to the Chairman of the AEC summarizing the views of ACRS on each reactor project is included in the appendix. During the 14 months covered by the report, reviews were completed on 9 reactors subject to licensing, 14 AEC-owned reactors, and 4 Department of Defense reactors. Additional information pertaining to the reviews of several of these reactors is discussed in the previous issue of this Review.

Experimental Gas-cooled Reactor Safeguards Review

Approval for the construction of the Experimental Gas-cooled Reactor (EGCR) at Oak Ridge has been granted by the Division of Licensing and Regulations, AEC. The EGCR is a power-demonstration reactor, with experimental facilities for testing large-scale experiments. The reactor is fueled with enriched uranium dioxide clad in type 304 stainless steel, moderated with AGOT graphite, and cooled with helium. The plant is designed to produce 85 Mw(t) and 25 Mw(e). The experimental facilities consist of eight straight-through loops in the reactor core; four of these loops will each be capable of handling 2 Mw(t), and the remaining

loops will each be capable of handling 0.5 Mw(t). The EGCR will be the first large-scale gas-cooled reactor to be built in the United States, and for this reason some of the hazards associated with this type of reactor are reviewed here.

The design of the EGCR and the accident and hazards analysis were presented recently in a hazards summary report prepared by Kaiser Engineers and Allis-Chalmers Manufacturing Company,[8] and certain points were further elaborated on in a supplement to the report.[9] The maximum credible reactor accident (mca) for the EGCR was considered to be the result of a rupture of the reactor coolant system. This accident would produce only one fuel-element cladding failure. The inlet and outlet air ducts to the containment shell would be closed on detection of the system rupture. At the time of the accident, it was assumed that one other fuel element that had failed might be present in the core. However, since there could be no assurance that the experimental loops would not burst following the loss of pressure in the reactor coolant system, all the experimental loops were assumed to be ruptured. Therefore the activity released to the containment shell would be that associated with two ruptured fuel elements plus the equilibrium activity associated with the eight experimental loops. Since the power generation associated with the two failed fuel elements would be approximately 29 kw and the power generation of the eight loops would be 10 Mw, the activity release from the ruptured fuel elements was neglected.

In the analysis of the consequences of the mca, it was assumed that 100 per cent of the volatile activity and 10 per cent of the nonvolatile activity was released to the containment shell or experimental loop cells. It was further assumed that all but 10 per cent of the released activity (except xenon and krypton) would be deposited within the containment shell, either by condensation or by removal in a recycle filter system within the containment vessel. Thus, in effect, only 10 per cent of the released activity, except xenon and krypton, would be available to leak out of the vessel.

The criterion for the EGCR design is that the mca will not result in an exposure to any individual on uncontrolled land of more than 25 r total body or equivalent in the first 8 hr following an accident. With this as the basis, it was established that the maximum leakage rate from the EGCR containment shell should be 2.2 per cent per day. One of the more important points that evolved in the course of the safeguards review pertained to the question of reduction of the gaseous contamination within the containment vessel following the mca, and the relation of such a reduction to the allowable leakage from the containment vessel. In response to the suggestion that the tightest attainable leakage specification be applied to the reactor containment vessel, regardless of the reactor activity release to the surroundings in question, the supplement to the EGCR preliminary hazards report[9] states:

> There does not appear to be any basis for accepting the complications and added costs attendant to an arbitrarily established leak rate which is lower than that shown to be necessary by a conservative analysis. Obviously, in any situation, additional safety is possible at greater expense and/or less performance requirements. The criterion for the EGCR design is that the maximum credible accident shall not result in an exposure to any individual on uncontrolled land of more than 25 r in the first eight hours following an accident. The resultant specification permits the containment-vessel leakage rate to be higher than that specified for many other containment systems. While it is true that it may be feasible to meet tighter specifications, it is not evident that the requirements of other systems should establish a precedent for the EGCR.

In support of this argument, a rough hazards index for a number of power-reactor containment systems was prepared[9] which was based on the containment-shell leakage rate, reactor power, and activity release. Although a more meaningful hazards index would include the distance from the reactor to off-site property, as well as the maximum allowable exposure, it was found that the hazards index for the EGCR was lower than that for any of the other reactors considered (Yankee, APPR, PWR, EBWR, and EBR-II). From such a comparison, it is apparent that the leakage rate for a particular reactor must be considered individually, based on the amount of activity which could be released and on the location of the reactor.

With reference to the credit taken for a reduction in the concentration of the activity within the containment vessel following the mca, it is pointed out in the supplement to the hazards summary report that there are several mechanisms for achieving this reduction,

including settling, plating out, filtering, and scrubbing. Although each of these natural processes will reduce the activity concentration, experimental data to support this contention are incomplete. Therefore a recirculation filter system is included in the EGCR containment system to assure that the activity release limits are maintained, even if the natural processes for activity removal are not effective. By maintaining a flow of 37,000 cfm through CWS filters backed up by silver-plated copper mesh (to collect iodine) and assuming a filter efficiency of 95 per cent, it can be shown that the total activity released from the containment vessel in the first 8 hr after the mca will be reduced by a factor of 12.

As noted previously, the hazards index for the EGCR compares favorably with the hazard indexes of all the water-cooled reactors. This is partly due to the reduction of the concentration of the nonvolatile activity within the containment vessel by the filtration system. Of equal importance, however, for the EGCR is the fact that the mca does not result in the release of appreciable amounts of activity from the fuel elements in the core. Therefore the activity release for the EGCR is due only to fission products resulting from 10 Mw of power generation instead of 95 Mw. Although this only results in a reduction in the potential activity release by a factor of 9.5, it is of far greater significance in a power reactor (i.e., for an EGCR with no experimental loops) in which the release would be due to 29 kw of power generation instead of 85,000 kw, or a factor of approximately 3000. The basis for the greatly reduced hazard in the gas-cooled reactor is the large heat capacity of the graphite. During the mca the fuel not only does not melt but also only one fuel element is likely to rupture.

A detailed analysis of the behavior of the fuel for the conditions of the mca is given in the hazards analysis,[8] in which the fuel-element cladding temperatures following rupture of the primary system were investigated as a function of time. The stress in the cladding was computed, based on the amount of fission gas present in the fuel for the exposures expected in the reactor. The number of fuel-element failures in the reactor was then computed on the basis of cladding temperature, which was determined from a statistical analysis of the various hot-channel factors and the pressure difference across the cladding which would cause the cladding to rupture at the elevated temperature. The results of this analysis depend, to a large extent, on the pressure build-up in the cladding, in which it was assumed that 50 per cent of the xenon and krypton would be released. Although this assumption is questionable on the basis of the present information available concerning the EGCR fuel elements, the number of fuel-element failures (even if the fission-gas release from the UO_2 were higher) would not be large compared with the number of failures that would exist in the case of a water-cooled reactor in which extensive melting would occur.

It should also be pointed out that a failure of a fuel element in the core of a gas-cooled reactor is not similar to the type of failure normally associated with the meltdown of fuel elements. The failures predicted for the EGCR fuel elements in the core at the time of the mca are cladding failures resulting from excessive pressure. The UO_2 temperatures would not be high enough to cause melting, and therefore the release of activity for the short time being considered would be much less than that experienced during meltdown. When these two factors are combined, it appears that the hazards index for a gas-cooled power reactor with no experimental loops may be sufficiently low that a containment vessel for controlling activity will be unnecessary. (H. N. Culver)

Safeguards Reports and Bibliographies

The bibliographies pertaining to hazards and safeguards of reactors and the safeguards reports which have recently been issued are listed below for reference. Because of the similarity of many reactors (in particular, the research reactors), the list is not intended to be all-inclusive.

Reports

1. J. H. MacMillan, Babcock and Wilcox Company, November 1958. (Unpublished.)

2. Gas-cooled Reactor Experiment No. 1 (GCRE-I) Hazards Summary Report, IDO-28506, December 1958, and Addendum, March 1959. (Confidential AEC report.)

3. D. S. St. John et al., Preliminary Hazards Evaluation of the Heavy-water Components

Test Reactor (HWCTR), DP-383, May 1959. (Unclassified AEC report.)

4. R. R. Eggleston and G. L. Schmidt, SNAP-II Experimental Reactor Hazards Report, NAA-SR-3465, April 1959. (Secret AEC report.)

Bibliographies

1. P. H. Wilks, The Release of Fission Products from Reactor Fuel; a Literature Survey, APEX-474, March 1959. (Secret AEC report.)

2. S. F. Lanier, Pressurized-water Reactors; a Literature Search, TID-3530, June 1959. (Unclassified AEC report.)

3. Proceedings of the 1958 AEC and Contractor Safety and Fire Protection Conference, Held at AEC Headquarters Building, Germantown, Maryland, June 24-25, 1958, TID-7569, May 1959. (Unclassified AEC report.)

4. R. J. Smith, Reactor Safety; a Literature Search, TID-3525, February 1959. (Unclassified AEC report.)

5. J. H. MacMillan, Babcock and Wilcox Company, June 1959. (Unpublished.)

6. Experimental Gas-cooled Reactor Preliminary Hazards Summary Report, ORO-196, May 1959. (This report is being revised; Unclassified AEC report.)

References

1. Lockland Aircraft Reactors Operations Office, AEC, Feb. 20, 1959. (Unpublished.)
2. General Electric Atomic Products Division, March 1959. (Unpublished.)
3. D. Callihan and J. T. Thomas, Accidental Radiation Excursion at the Y-12 Plant. I. Description and Physics of the Accident, *Health Physics*, 1(4): 363-372 (1959).
4. J. D. McLendon, Accidental Radiation Excursion at the Y-12 Plant. II. Health Physics Aspects of the Accident, *Health Physics*, 2(1): 21-29 (1959).
5. Accidental Radiation Excursion at the Y-12 Plant, June 16, 1958, Y-1234, July 28, 1958. (Unclassified AEC report.)
6. Marshall Brucer, comp., The Acute Radiation Syndrome; A Medical Report on the Y-12 Accident, June 16, 1958, ORINS-25, April 1959.
7. Indemnification of Atomic Energy Activities and Operation of the Advisory Committee of Reactor Safeguards, Report to the Joint Committee on Atomic Energy, USAEC, Mar. 31, 1959.
8. Experimental Gas-cooled Reactor Preliminary Hazards Summary Report, ORO-196, May 1959. (This report is being revised; Unclassified AEC report.)
9. Experimental Gas-cooled Reactor Preliminary Hazards Summary Report, ORO-196 (Supplement), July 17, 1959. (This report is being revised; Unclassified AEC report.)

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